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Representation of the Few-Group Homogenized Cross Sections of a MOX Fuel Assembly

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Abstract content
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Nodal diffusion methods are often used to calculate the distribution of neutrons in a nuclear reactor core. They require few-group homogenized neutron cross sections for every heterogeneous sub-region of the core. The homogenized cross sections are pre-calculated at various reactor states and represented in a way that facilitates the reconstruction of cross sections at other possible states. In this study a number of such representations were built for the cross sections of a MOX (mixed oxide) fuel assembly via hierarchical Lagrange interpolation on Clenshaw-Curtis sparse grids. Traditionally, nodal reactor core simulators have employed cross sections with two energy groups, but there is evidence that more energy groups are needed to simulate reactor cores that contain MOX. Representations were therefore constructed for both the traditional two energy groups and a six energy group structure. Both the rate at which the representation accuracy improves with the number of samples and the complexity of the cross section dependence on individual state parameters were examined. The anisotropy feature of the representation procedure, which allows more samples to be taken for state parameters that are known to be more important to the representation accuracy than others, was applied throughout. The results show that the representation method allows both two-group and six-group cross sections to be represented in a computationally efficient manner to an industrially acceptable level of accuracy, despite additional complexity in the dependence of six-group cross sections on the state parameters.

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