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Burnup Analysis Capability Based on Unstructured Mesh Geometry in Reactor Monte Carlo code RMC

摘要

非结构网格（Unstructured Mesh, UM）是蒙卡（Monte Carlo, MC）程序一种先进的几何表述方式，在复杂几何建模、精细计数统计、高保真多物理耦合等方面具有天然优势。尽管在基于 UM 几何的粒子输运方面已有显著研究进展，但对于反应堆全寿期的模拟还需要具备燃耗计算分析能力，MC 程序直接基于 UM 几何的燃耗计算方法仍有待进一步研究。本文聚焦于 MC 程序在 UM 几何框架下的燃耗计算方法研究，在 RMC UM 几何下粒子输运和计数统计能力的基础上，进一步实现了 UM 几何下的燃耗计算功能。为 RMC 建立起 KRUSTY 反应堆的 CSG 和 UM 燃耗计算模型，结果表明，UM 与 CSG 模型在各个燃耗步下的 keff 及重要核素质量变化吻合很好，验证了 RMC 基于 UM 几何进行燃耗计算功能的正确性。

关键词

蒙卡程序；非结构网格；燃耗计算；RMC

Abstract

Unstructured Mesh (UM) serves as an advanced geometric modeling method in Monte Carlo (MC) codes, offering inherent advantages for representing complex geometries, obtaining high-resolution tally results, and enabling high-fidelity multi-physics coupling. Although significant progress has been made in UM-based particle transport simulation within MC codes, full lifecycle reactor simulation still requires burnup calculation and analysis capabilities. The development of burnup calculation methods based on UM geometry in MC codes remains an area requiring further investigation. This study focuses on burnup calculation methodologies under the UM framework in MC codes. Leveraging the particle transport and tally capabilities of RMC's UM geometry, we have further implemented burnup calculation functionality based on UM in RMC. Both CSG and UM burnup calculation models of the KRUSTY are established for RMC. Results demonstrate consistent agreement between the UM and CSG models in terms of keff and the evolution of key nuclide masses across different burnup time steps, confirming the accuracy of RMC's UM-based burnup calculation capability.

Keywords

Monte Carlo code; Unstructured Mesh; Burnup Calculation; RMC

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