

先进核反应堆高保真中子学计算方法与应用研究进展

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摘要 西安交通大学核工程计算物理实验室(NECP团队)20余年来面向国家对于先进核能与核动力的重大需求, 围绕先进核反应堆中子学分析开展了系统的高保真中子学计算方法与应用研究. 本文首先简要介绍了中子学分析涉及的关键理论方法, 包括核数据加工、共振计算、输运计算、均匀化、多物理耦合方法; 然后介绍了基于理论方法自主研发的反应堆中子学分析系列核心软件, 包括核数据制作软件NECP-Atlas、压水堆物理分析软件NECP-Bamboo系列、数值反应堆软件NECP-X、快堆分析软件NECP-SARAX、高温气冷堆分析软件NECP-Panda、屏蔽分析软件NECP-Hydra和NECP-MCX等; 最后选取了几个典型案例介绍了所开发的软件在先进核反应堆设计与核电运行支持方面的应用.

关键词 核反应堆物理, 中子学计算, 核数据加工, 压水堆, 快堆, 高温气冷堆

核反应堆技术是世界各国竞相发展的核心技术, 其堆芯设计与安全分析都依赖于反应堆中子学计算分析. 随着反应堆技术不断升级, 长期以来国际上一直沿用的基于各种近似假设的中子学计算方法已无法满足先进反应堆精准分析的需求. 西安交通大学核工程计算物理实验室(NECP团队)20余年来紧密围绕先进核反应堆研发的需求, 进行了系统的“高精度、高精度、高置信”高保真中子学计算方法研究, 自主研发了先进核反应堆中子学分析系列核心软件, 并成功应用于我国各种堆型的堆芯设计与运维支持, 实现了核心技术自主可控, 显著提升了我国先进核反应堆的研发能力和核电机组的运行水平.

1 先进核反应堆高保真中子学计算方法

中子学计算的基础是核数据库, 若采用一步法进行堆芯中子学计算, 需要在核数据的基础上进行堆芯规模的共振计算和输运计算; 若采用两步法, 需要在组

件共振计算和输运计算的基础上进行均匀化, 然后开展堆芯扩散计算. 另外, 为了模拟核反应堆内的真实状态, 需要进行中子场、温度场、流场、应力-应变场等多个物理场的耦合计算. 以下将分别介绍已开展的相关工作.

1.1 堆用核数据库制作方法

核数据库是中子学分析的根基, 堆用核数据库是通过评价核数据库进行加工制作后形成的、可直接用于反应堆中子学分析的专用核数据库(也称工作库), 包括多群数据库和连续能量数据库两种形式. 核数据库制作主要包括共振重构、多普勒展宽、热散射处理、多群归并等环节, 其详细过程见文献[1]. NECP团队在传统制作方法的基础上, 针对慢化材料热散射律数据精度低、核数据引起的中子学分析不确定性复杂等问题开展了研究, 提出了热散射律数据评价方法、核数据不确定度分析及同化方法, 完善了堆用核数据

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库制作方法.

1.1.1 热散射律数据评价方法

热散射律数据是描述热中子与材料散射后次级中子能量、角度分布的基础数据. 我国缺少自主化的热散射律数据, 国外最新的热散射律数据库虽然包含了一定数量材料的热散射律数据, 但是, 一些重要材料(例如: 石墨、铍等)仍是采用传统方法进行评价, 数据精度低, 无法满足新型核反应堆高精度计算的需求. NECP团队针对晶体、液体材料分别建立了热散射律数据评价方法: 基于第一性原理建立了晶体材料热散射律数据计算方法, 提出了直接方法、各向异性位移参数方法^[2]、一声子展开方法等先进方法^[3], 消除了传统热散射律数据计算中的近似, 提高了相干弹性散射、非弹性散射数据的计算精度.

1.1.2 核数据不确定度分析及同化方法

评价核数据不可避免存在不确定度, 不确定度会通过中子学计算过程传递给最终结果. 新型核反应堆的材料成分和能谱特点与传统反应堆显著不同, 涉及的重要核数据的不确定度大于传统反应堆. NECP团队基于抽样方法建立了不确定度分析方法: 首先, 建立了针对评价核数据库的抽样方法^[4], 可以实现对评价核数据库中所有数据精确抽样, 拓展了不确定度分析的数据类型, 提高了不确定度分析的精度; 其次, 建立了基于多群协方差堆用核数据库抽样方法^[5], 根据用户的数据库格式实现不确定度量化. 在不确定度分析的基础上, 基于广义最小二乘、贝叶斯推理、贝叶斯-蒙特卡罗等方法建立了核数据同化方法, 根据核反应堆宏观测量结果对核数据进行优化, 提高反应堆中子学计算的精度^[6].

1.2 共振参数计算方法

中子与铀、钚等原子核相互作用的核反应截面在部分能量区间存在剧烈的共振峰. 应用确定论方法求解中子输运方程前, 需要通过共振计算获得铀、钚等共振核素共振能量区间的有效共振自屏截面, 因此共振计算的精度直接决定了确定论中子输运计算方法的精度, 对核反应堆的设计和安全分析至关重要. 然而, 共振自屏计算与核燃料的成分、几何、温度等因素紧密相关, 加上共振峰存在干涉效应和自屏效应, 计算异常复杂, 长期以来人们均采用各种近似方法, 共振参数计算的精度和适用性问题已成为制约先进反应堆中子学分析的瓶颈. NECP团队针对多核素干涉效应, 提出

了伪核素子群方法^[7]、小波函数展开法^[8]、小波-子群耦合共振计算方法^[9]、基于深度学习^[10]的共振计算方法等, 建立了适用各种能谱特点的核反应堆共振计算方法体系, 并针对商用压水堆、弥散颗粒燃料压水堆、快中子反应堆等堆型提出了有效的共振自屏计算方案.

1.2.1 商用压水堆共振计算方法

为了进一步提高压水堆全堆芯共振计算的效率, 提出了全局中子流-局部伪核素子群的耦合共振计算方法^[11], 以及全局嵌入式自屏和局部超细群的耦合共振计算方法^[12]; 由于局部伪核素子群方法和局部超细群方法主要针对圆柱或平板几何, 为了拓展全局-局部耦合共振计算方法的几何适用性, 提出了基于逃脱概率守恒的等效几何共振计算方法^[13], 可将全局-局部耦合共振计算方法拓展至复杂几何燃料的全堆芯共振计算, 从而解决燃料形状、干涉效应等给压水堆燃料有效自屏截面计算带来的难题.

1.2.2 弥散颗粒燃料压水堆共振计算方法

与常用的均匀二氧化铀陶瓷燃料相比, 弥散颗粒燃料具有更高的裂变产物包容能力、导热性能和辐照稳定性. 相比二氧化铀等均匀燃料, 燃料芯体中颗粒和基体间的微观非均匀性, 以及燃料棒间的宏观非均匀性形成了双重非均匀性系统, 这对装载弥散颗粒燃料的核反应堆的共振计算方法提出了巨大的挑战. NECP团队研究了基于一维球模型的缺陷因子方法^[14,15], 实现了弥散颗粒燃料的共振计算. 为了提高计算精度, 提出了基于颗粒丹可夫因子守恒的改进缺陷因子方法^[16]. 为了进一步处理包含多种类型颗粒的问题, 提出了基于Hébert模型的全局-局部耦合共振计算方法^[17], 实现弥散颗粒燃料的高精度共振计算.

1.2.3 快中子反应堆共振计算方法

快中子反应堆相比压水堆等热中子反应堆能谱更硬, 共振核素更多, 其共振计算方法与热中子反应堆具有显著的差异. NECP团队研究了可同时考虑堆芯燃料、冷却剂和结构材料共振效应的TONE方法^[18], 研究了不可分辨共振自屏计算模型^[19]. 为了提高共振计算方法的能谱适用性, 进一步研究了基于点截面和超细群的全能量范围超细群计算方法^[20], 不但可以进行快中子反应堆的共振计算, 对于宽能谱反应堆的共振计算同样适用.

1.3 中子输运计算方法

通过确定论或概率论数值方法求解中子输运方程

是开展核反应堆设计和辐射场分析的基础。传统方法基于几何均匀化思想,把复杂几何等效成规则几何(主要是正方形和六角形)进行中子输运计算,不可避免地引入了几何近似,面对先进核反应堆更加复杂的几何,计算精度受到挑战。NECP团队在非结构几何确定论中子输运数值方法和高效概率论蒙特卡罗输运数值方法等方面开展了系统性的研究,攻克了复杂几何先进核反应堆的中子输运计算难题。

1.3.1 基于三角形网格的确定论中子输运计算数值方法

采用三角形网格对模拟区域进行网格离散,可以实现对核反应堆复杂几何的高精度建模。然而,对于空间-角度-能量强相关的一阶中子输运方程,传统基于三角形网格的有限元方法面临数值不稳定、内存消耗过大、迭代收敛困难等难题。针对核反应堆中子输运计算的具体特征,发展了SAAF形式中子输运方程的有限元方法^[21]、一阶最小二乘有限元方法^[22]、一阶双曲型方程连续有限元稳定算法^[23]等有限元改进算法,提出了三角形网格穿透概率方法^[24]、三棱柱网格离散纵标节块法^[25]、三棱柱网格中子流方法^[26]、三棱柱网格SP₃节块方法^[27]等新方法。

1.3.2 基于特征线的确定论中子输运数值方法

特征线方法被认为是具有任意复杂几何建模能力的确定论中子输运计算数值方法,是核反应堆高保真计算的主要方法。然而,传统特征线方法面临特征线信息产生困难、存储量巨大、计算时间过长等难题。NECP团队建立了基于AutoCAD的二维任意几何特征线方法^[28]、基于矩阵特征线的二维/一维耦合输运计算方法^[29]、CPU-GPU异构并行的特征线计算方法^[30]、基于材料区的二维一维耦合特征线方法^[31]和特征线在线生成的特征线方法^[32],提出了基于2D MOC/3D SN的耦合计算方法^[33]以及三维模块化特征线方法^[34]等多种新方法。

1.3.3 基于概率抽样的蒙特卡罗中子输运数值方法

蒙特卡罗方法通过抽取大量的样本完成对中子输运过程的数值模拟,理论上不受几何复杂度的影响。但是,在大规模、深穿透的中子输运计算中,蒙特卡罗方法存在假收敛、深穿透区域计数难和计算资源消耗过大等难题。NECP团队建立了一致性共轭驱动重要性抽样(CADIS)方法^[35],提出了自适应网格权窗产生技术^[36]、基于DXTRAN的权窗产生技术^[37]、CADIS-下次事件估计器耦合算法、首次碰撞源-CADIS耦合算

法、CADIS-DXTRAN耦合算法^[38]等新方法。相比传统蒙特卡罗方法,该方法显著降低了深穿透区域的方差,极大提升了数值模拟的效率。

1.4 均匀化计算方法

长期以来,核反应堆堆芯中子学计算一般采用基于均匀化理论的两步法。均匀化并非简单地打混,需要保证特定物理量在均匀化前后的守恒性,包括核反应率、中子泄漏率、有效增殖因子、缓发中子产生率、衰变热释放率等。NECP团队针对传统的均匀化方法提出了诸多改进措施,显著提升了计算精度。

1.4.1 压水堆堆芯均匀化计算方法

压水堆堆芯内的中子学特点是局部非均匀、整体均匀。NECP团队提出了基于非均匀组件一步法计算的组件均匀化计算方法^[39]、面向小型堆或重反射层结构或六角形几何的二维反射层均匀化计算方法^[40,41]、处理环境效应的子网格均匀化方法^[42]、面向各向同性和各向异性SP₃理论的光薄栅元均匀化方法^[43~46]、考虑中子泄漏对中子能谱影响的非均匀泄漏修正理论^[47]、能量维度上从子群到能群的归并均匀化方法^[48]、核素维度基于定量重要性分析的燃料链压缩方法^[49,50]等,显著降低了均匀化处理带来的误差;还提出了基于P₁理论的压水堆堆芯物理计算方法,针对pin-by-pin计算,已实现以扩散理论的计算代价获得与SP₃理论相当甚至更优的计算精度^[51]。

1.4.2 快堆堆芯均匀化计算方法

快堆具有中子能谱硬、中子平均自由程长、燃料组分复杂、共振效应剧烈等特点。NECP团队提出了考虑截面角度相关性的均匀化参数计算技术^[52]、自适应能群结构产生技术^[53]、基于等效堆芯的能群等效压缩技术^[54]、高能数据均匀化参数计算技术以及中子、光子释热参数产生技术^[55]等,进一步开展了含高能数据的均匀化参数计算技术以及基于平均宏观泄漏截面的在线均匀化技术等^[56,57],显著降低了计算误差。

1.4.3 高温气冷堆堆芯均匀化计算方法

与传统的压水堆相比,球床式高温气冷堆存在燃料双重非均匀性、连续在线换料、球床堆芯结构等特点。球床式高温气冷堆的堆芯具有瘦长几何、超热能谱的特点,活性区中子泄漏对堆芯能谱影响显著。NECP团队采用基于蒙特卡罗精细建模的在线中子泄漏修正方法获得精确少群常数^[58],在均匀化截面计算、控制棒区域中子流效应修正等理论方法上进行了改进,

实现了高精度堆芯中子学计算。

1.5 多物理耦合计算方法

除了中子场,反应堆的运行状态还受其他物理场的影响,如温度场、流场、应力-应变场等,且各个物理场之间相互影响。因此,需要进行中子学、热工水力与燃料性能之间的多物理耦合计算分析。

针对稳态分析,NECP团队开展了基于自适应松弛的高保真中子学-热工水力-燃料性能多物理耦合计算方法研究^[59],建立了基于Picard迭代的多物理耦合方程求解方法、物理场网格映射方法及针对大规模并行的数据传递方法;针对传统耦合松弛方法的松弛因子依赖人工经验给定的问题,提出了基于迭代历史的自适应松弛方法,相比传统松弛方法,该方法消除了松弛因子选取对人工经验的依赖,且具有很好的计算鲁棒性。

针对瞬态分析,开展了基于JFNK迭代法的堆芯瞬态核热耦合计算方法研究^[60]。针对核、热不同方程间自变量存在量级差异问题,通过引入尺度变换对核热耦合方程进行同解变换,保证了牛顿迭代的快速收敛。此外,核热耦合方程的雅可比矩阵是一个高度病态的矩阵,利用基于Picard方法的核热耦合求解器对矩阵进行预处理,将牛顿内迭代次数降低了两个量级。进一步研究了基于JFNK的中子学方程、热工水力学方程和燃料力学方程(核-热-力)的耦合求解方法,可精确考虑瞬态过程中燃料热膨胀对燃料棒中心温度的影响。

2 先进核反应堆物理分析软件研发

基于上述理论方法和发明的配套技术,自主研发了具有完全自主知识产权的高保真中子学分析软件系统,并经过了大量实测数据的验证与确认。

2.1 堆用核数据库制作软件

团队研发了堆用核数据库制作软件NECP-Atlas^[1],软件具有完善的核数据处理功能。NECP-Atlas支持传统的ENDF-6格式评价核数据库的处理,完成了针对中国的CENDL、美国的ENDF/B、经合组织核能署(OECD/NEA)的JEFF、日本的JENDL、欧洲的TENDL评价核数据库的测试验证,成功将中国最新版CENDL-3.2应用到核反应堆堆芯设计,实现了反应堆设计核数据库的国产化。同时,NECP-Atlas还支持目前国际上推广的GNDS格式评价核数据库,在国际上率先将GNDS格式数据库应用到核反应堆物理、屏蔽、不

确定量化等分析^[61]。

NECP-Atlas软件可处理评价核数据库中提供的中子核反应、光子-原子反应、光子-原子核反应、原子核衰变、裂变产额等数据,制作产生中子-光子输运计算所需的反应截面及次级粒子能量-角度分布数据^[62]、燃料计算所需的燃料链数据^[63]、活化源项计算所需的次级粒子发射数据、材料辐照损伤计算所需的离位损伤截面^[64]、能量沉积计算所需的释热因子^[65]、不确定度分析所需的协方差数据^[66]等。

图1给出了采用NECP-Atlas计算的石墨的总截面数据与基于国际上ENDF/B-VIII.0评价核数据计算的总截面以及实验结果的对比,可见,NECP-Atlas的结果与实验吻合度更高,相对于ENDF/B-VIII.0最大改进28.6%。

2.2 压水堆堆芯物理分析软件

研发了先进压水堆堆芯物理分析软件Bamboo-C^[67~72],核心功能程序包括组件程序LOCUST、堆芯程序SPARK和参数化程序LtoS。Bamboo-C软件已基于压水堆核电厂近300堆年运行数据完成了工业确认,确认对象覆盖了我国商用压水堆所有堆型,见图2,包括“国和一号”、“华龙一号”、AP1000、EPR1750、VVER1000、M310及其改进型、CNP650和CNP300,确认内容涵盖压水堆启动物理试验和功率运行中堆芯关键安全参数,包括临界硼浓度、温度系数、控制棒价值、三维功率分布等。工业确认结果表明:Bamboo-C软件对堆芯关键安全参数的计算精度均满足限值要求,具备压水堆核电厂工业应用所需的分析能力。目前,该软件已通过生态环境部核与辐射安全中心的核安全评价,成为国内首个国产审评软件,可用于核安全审评分析。

研发了基于栅元均匀化的物理分析软件NECP-Bamboo2.0^[73~77],与采用粗网扩散方法的现有工业软件不同,该软件采用pin-by-pin两步法,直接在三维全堆芯范围内以栅元尺度进行中子输运、水力流动、热量传递、核素燃料、探测器响应等多物理耦合计算^[77~79],能适配银河麒麟等国产操作系统。软件已通过12种型号商用压水堆和4种板元件实验堆的实测数据验证与确认^[80~82],已经过15个反应堆共83个样本的辐照后实验(PIE)核素成分测量数据的验证与确认^[83]。图3给出了典型压水堆堆芯计算结果误差对比。

基于全堆芯一步法的高保真数值反应堆堆芯物理

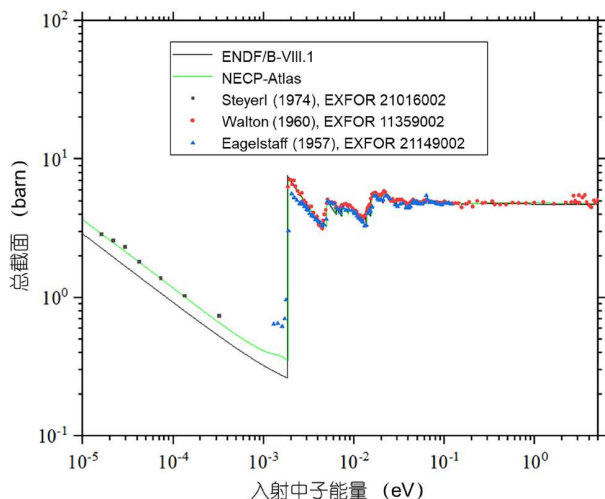


图1 296 K石墨的热散射截面

Figure 1 Thermal scattering cross section of graphite at 296 K

分析软件NECP-X^[84~88], 采用CSG几何实现了复杂几何堆芯的精细建模能力。程序具备全堆芯稳态、瞬态、循环燃耗和核-热-燃料性能多物理耦合高保真计算等功能, 可在单台服务器和大型超算平台进行灵活部署和高效运行。利用单台服务器, 平均1.5 min可完成M310机组1/4堆芯单个燃耗点的计算, 克服了只能在超算平台使用高保真数值反应堆技术的瓶颈。已利用M310、CPR1000、AP1000、EPR等100多个堆年的商用压水堆实测数据进行了测试验证, 并为“华龙一号”、“国和一号”、EPR等先进大型压水堆的设计验证

和堆芯运行提供了技术支持。利用SPERT、JRR-3、西安脉冲堆、高通量堆等实验堆对程序在小型反应堆计算方面进行了深入的测试验证, 并应用于上海核工程设计院、中广核研究院的棒状、板状和异形燃料等小型动力堆, 克服高泄漏、强非均匀性小型反应堆难以精确计算的难题, 为小型反应堆堆芯设计提供参考结果。图4给出了NECP-X的多物理耦合策略和可视化效果。

2.3 快堆物理分析软件

研发了快堆分析软件NECP-SARAX^[89~92], 该软件基于两步法框架, 采用了连续能量点截面的超精细能谱计算、三维非结构几何并行中子-光子输运计算、三维网格核-热-力耦合计算等一系列新方法, 与数学物理模型, 解决了快堆所面临的复杂能谱计算、复杂几何建模等问题, 实现了高精度的快堆物理计算分析, 可用于快谱堆芯的临界、燃耗、反应性、动力学参数计算和堆芯燃料管理、堆芯瞬态过程模拟与多物理耦合分析, 也可拓展到超热谱及热谱堆芯的应用。

NECP-SARAX软件核心模块包括: 能谱与截面计算模块TULIP、稳态堆芯计算模块LAVENDER以及瞬态堆芯分析模块DAISY。该软件通过了国际临界基准例题、反应堆物理基准实验等数据库中提供的20余个装置百余套基准实验数据的验证与确认, 并在国内通过了铅铋零功率装置^[93]、中国实验快堆^[94]、中国示范快堆启动物理试验^[55,95]等的实堆实测检验, 计算精度

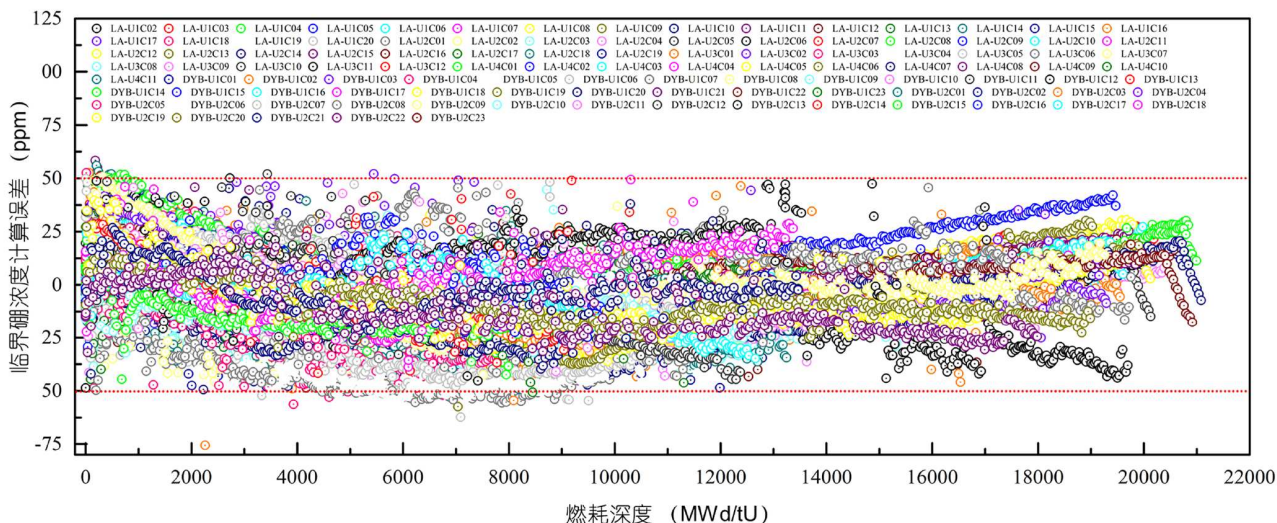


图2 Bamboo-C软件验证结果

Figure 2 Bamboo-C code and validation results

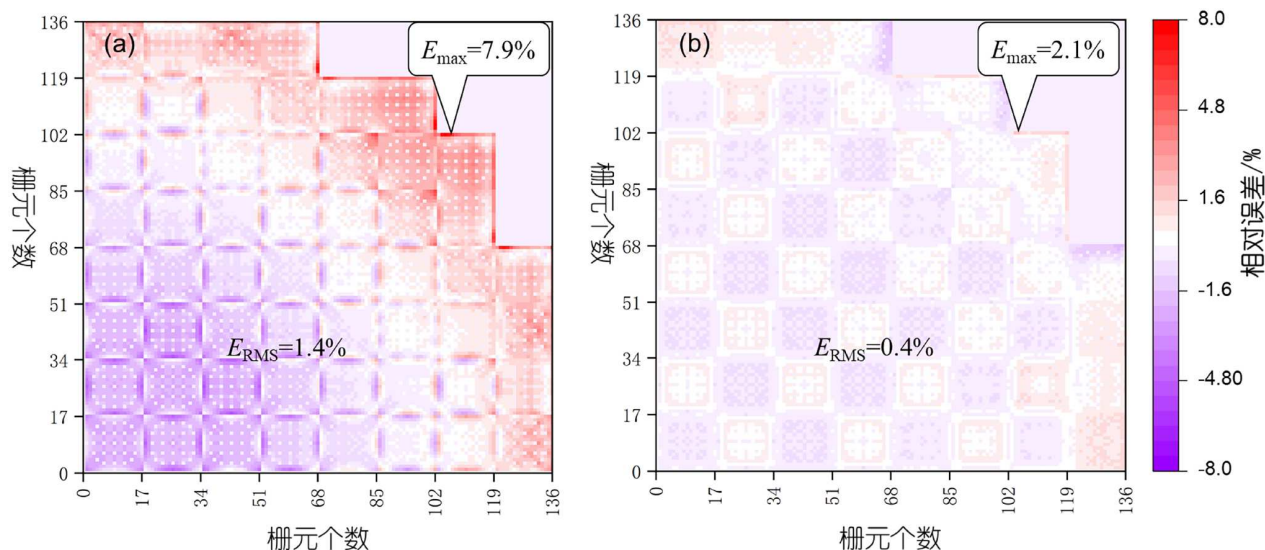


图3 压水堆堆芯的燃料棒功率计算相对误差。(a) 粗网扩散方法; (b) 细网pin-by-pin方法

Figure 3 Relative error of pin power in the PWR core of the coarse-mesh diffusion method (a) and the fine-mesh pin-by-pin method (b)

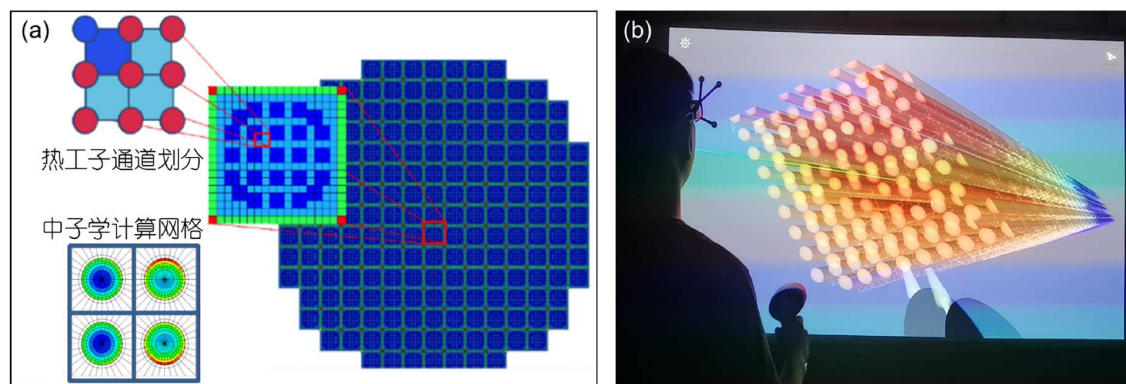


图4 NECP-X程序的多物理耦合策略(a)和三维可视化效果(b)

Figure 4 Multi-physics coupling strategy (a) and 3D visualization (b) of the NECP-X code

得到了系统且全面的确认, 满足工程设计要求. 通过IAEA组织的20余个研究机构的钠冷快堆程序对标^[96], NECP-SARAX软件系统计算精度达到或优于国外主要国家专用快堆软件的精度, 见图5.

2.4 高温气冷堆分析软件

高温气冷堆采用球床式设计, 堆芯包括数十万个石墨基体燃料球, 每个燃料球中包括大量的TRISO颗粒燃料, 具有显著的双重非均匀性. 为了进一步提升球床式高温气冷堆堆芯物理计算精度, NECP团队在充分吸收VSOP、PANGU等程序的理论方法和经验的基础上, 进一步改进形成了一套优化的高温气冷堆堆芯物

理计算方法, 自主开发了NECP-Panda软件. 相比现有方法, 对均匀化截面计算、控制棒区域中子流效应修正、TRISO燃料颗粒温度计算等理论方法进行了改进^[97~99], 提升了软件的计算精度. 基于我国石岛湾示范工程项目HTR-PM的验证结果表明, 软件能够准确计算吸收体价值、温度反应性、毒物反应性、堆芯温度分布等关键物理热工参数, 证明其可用于球床式高温气冷堆的设计分析和运行预测, 如表1所示.

2.5 屏蔽分析软件

研发了基于离散纵标方法的辐射场计算软件NECP-Hydra, 可实现大型压水堆核岛厂房、核动力系

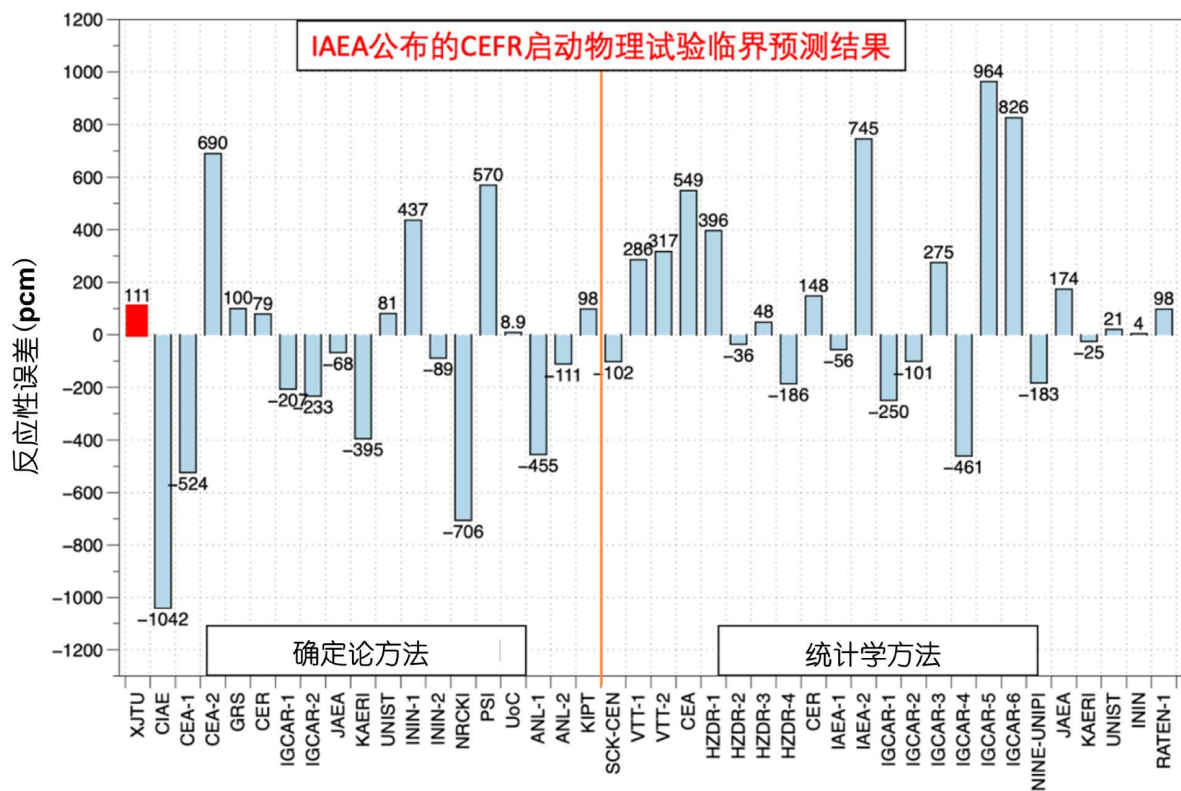


图 5 快堆物理分析软件NECP-SARAX模型及验证结果
Figure 5 Model and validation results of fast reactor physics analysis code NECP-SARAX

表 1 高温气冷堆分析软件NECP-Panda的数值结果
Table 1 Modeling and numerical results of high-temperature gas-cooled reactor analysis code NECP-Panda

混合球装载高度 (cm)	参考解	NECP-Panda	Keff偏差(pcm)
220	0.94891	0.94817	-74
250	0.98126	0.98197	68
275	1.00388	1.00430	42
300	1.02290	1.02266	-24
330	1.03953	1.04071	118
385	1.06522	1.06567	46
440	1.08333	1.08369	36
495	1.09836	1.09740	-96

统等大规模问题三维一次性输运计算。该软件采用改进KBA并行算法、C++及FORTRAN混合编程、优化的程序框架和数据结构设计，显著提升软件运行性能，万核并行效率高于50%。软件已在压水堆一次侧屏蔽计算、CFR600堆本体中子场计算、乏燃料贮藏罐计算、AP1000安全壳全厂房屏蔽计算(如图6所示)、海洋核动力系统、多种新型反应堆概念设计的屏蔽计算

等实现成功应用，在方案设计、型号研发、科学研究等方面发挥了重要作用。

研发了基于蒙特卡罗-确定论耦合方法的辐射场计算软件NECP-MCX，具备中子-光子耦合输运计算^[100]、非结构网格粒子输运^[101]、人体剂量计算^[102]、输运-燃料-活化-源项耦合计算^[103]、连续能量核数据敏感性分析^[104,105]、中子噪声^[106]、点核积分^[107]、多群计算、随机弥散问题模拟^[108]、群常数产生^[108]、几何建模可视化和计算结果可视化等功能。软件经过大量的基准题验证，与国内外同类软件相比，NECP-MCX在保证高精度的基础上，对于屏蔽深穿透问题具有更高的计算效率。对于辐照监督管响应的计算，对比传统方法，NECP-MCX采用的概率论-确定论耦合方法可将品质因子(FOM)提高100倍以上^[35]。随着CPU数目的增加，传统方法并行效率急剧下降，千核并行效率低于50%，而NECP-MCX的千核并行效率保持在90%^[35]。软件已应用于“华龙一号”、AP1000、M310、示范快堆等堆型的堆外探测器响应模拟，数值模拟结果与实测结果吻合良好；应用于中国聚变工程实验堆的中子学设计、

装备的辐射屏蔽设计、西安脉冲堆辐照孔道屏蔽设计等, 计算精度与国外同类软件相当, 计算效率高于同类软件(见图7)。

经第三方独立测试, NECP-Hydra软件成熟度达到工业软件的8.4级, 质量评价为“优秀”, 自主可控比例超过90%, 自主可控测评达到A级, 属于完全自主可控; NECP-MCX软件成熟度达到工业软件的8.2级, 质量评价为“优秀”, 自主可控比例超过90%, 自主可控测评达到A级, 属于完全自主可控。两款软件均入选了国家国防科技工业局印发的《工业软件自主可控产品目录(2023版)》。

3 先进反应堆设计与运行应用技术

上述软件系统被团队和合作单位广泛应用于各种堆型, 在此基础上, NECP团队针对新型核反应堆设计以及核电厂运行的工程需求开展了多项支持技术的研究, 以更好地服务于工程应用。

3.1 新型反应堆设计

基于反应堆物理分析程序SARAX开展了第四代核能系统代表堆型钠冷快堆, 以及铅铋小堆等新型核反应堆的设计和支持技术研究。

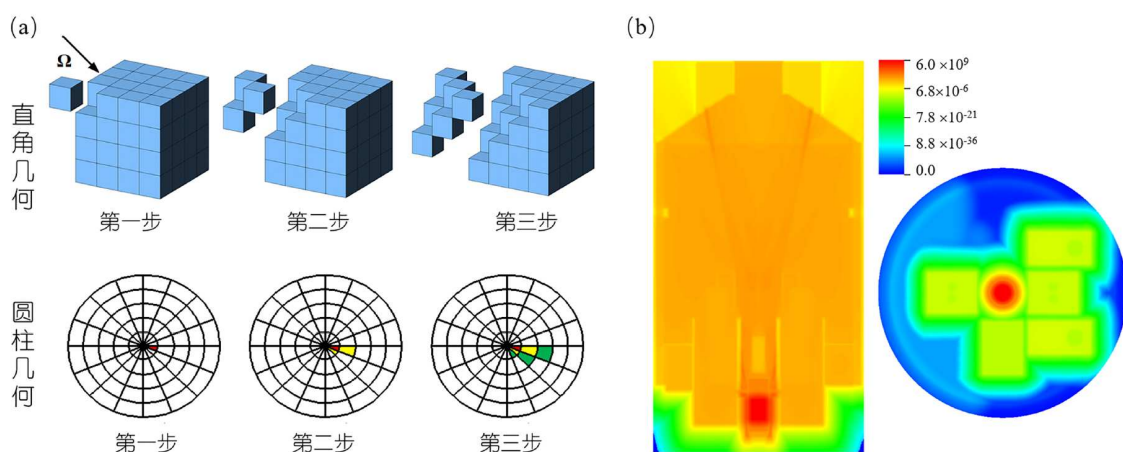


图6 NECP-Hydra软件理论模型和数值结果。(a) 程序的KBA并行扫描算法; (b) AP1000核岛厂房快中子通量分布计算结果

Figure 6 Theoretical model and numerical results of NECP-Hydra. (a) The KBA parallel sweep strategy; (b) the calculation results of fast neutron distribution of AP1000

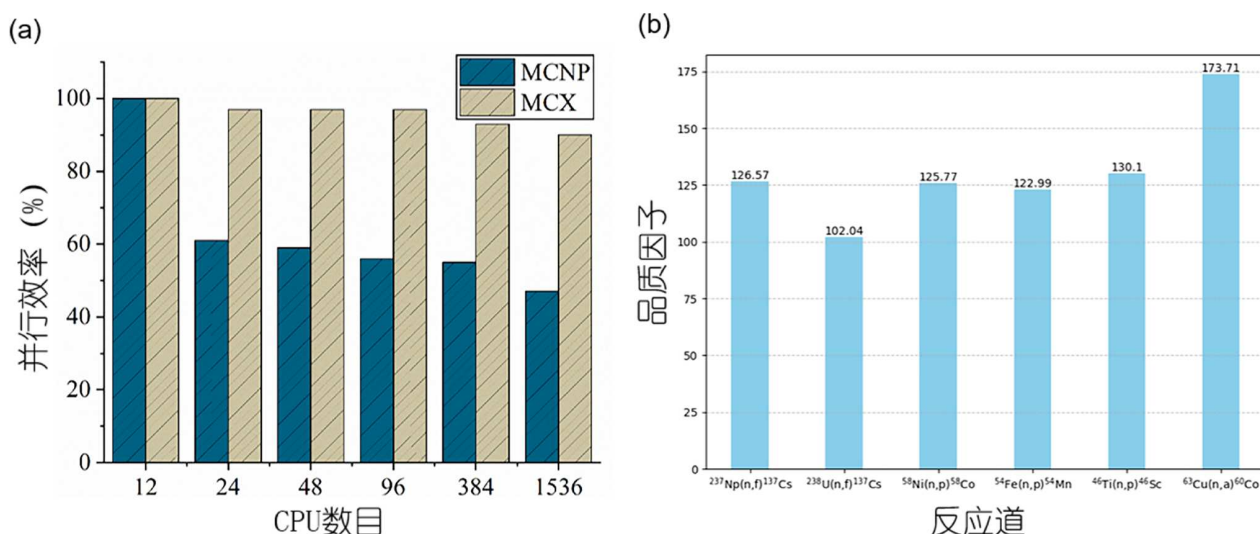


图7 NECP-MCX程序的并行效率(a)和品质因子(b)

Figure 7 Parallel efficiency (a) and merit (b) of the NECP-MCX code

3.1.1 微小型反应堆在线能谱修正技术

为实现反应堆小型化,某些特殊应用场景的反应堆设计方案通常采用以快谱堆芯为核心,辅具有强化能力的反射层以减小堆芯体积。这一特殊设计使得反应堆堆芯从原有单一能谱特征演变为从堆芯内部到外部区域跨越快谱、中能谱、超热谱、热谱的复杂能谱特征,传统核设计技术与软件无法满足精度要求。为此,提出在连续能量点精细能谱计算基础上,采用堆芯在线能谱修正技术进行三维堆芯中子输运计算,从而解决了确定论两步法核设计计算中堆芯不同区域能谱差异导致的计算误差大的问题。该技术应用于国内某重点型号微型反应堆的核设计,将关键核参数的设计计算误差从原有超过1500 pcm降低到核安全许可的500 pcm以下,为型号研发发挥了重要的作用。

3.1.2 三维大范围高效率中子输运分析技术

受高温、高辐照等严苛环境的影响,大型池式钠冷快堆的堆外核测系统在远离堆芯活性区的位置,中子经专设的引出通道输运到堆外核测系统处,信号衰减超过10个数量级,且存在强烈的各向异性特征。这些复杂的物理效应使得传统的中子输运数值分析技术在进行数值模拟时会导致不可接受的计算误差,超出核安全允许的范围。为此,提出了针对快中子反应堆的三维大范围高效率中子输运分析技术,可以实现万亿自由度(包括空间网格、能群、角度)的超大范围中子场直接数值模拟,给出大型池式钠冷快堆三维高精度的中子注量率分布。该技术被应用于国家重大核能工程,复核了大型池式钠冷快堆首堆首次物理启动的堆外探测器标定试验,给出了最接近试验测量数据的数值模拟结果,对保障重大工程按期投入运行发挥了重要的支撑作用。

3.2 核电站反应堆运行支持

为了进一步提升我国压水堆核电厂运行的安全性和经济性,基于NECP-Bamboo-C软件自主研发了多项先进的核电站反应堆运行支持技术,并且在压水堆核电厂实现了工程验证和应用,主要包括:动态通量测量技术、单点校刻技术、改进动态刻棒技术、次临界刻棒技术等。

3.2.1 动态通量测量技术

压水堆机组每次停堆换料大修后,核安全法规要求必须在机组升功率过程中典型的30%FP、75%FP和100%FP等功率台阶实施堆芯通量测量试验,以保障机

组的安全性。传统平衡氙通量测量技术要求在每个功率台阶等待超过24小时、氙毒达到平衡之后再实施测量试验,显著影响了机组的经济性。动态通量测量技术依赖Bamboo-C软件的高精度和可靠性,通过对氙毒分布时空特性的高精度数值模拟,基于动态氙条件下堆芯通量测量试验的实测值,能够高精度地预测平衡氙条件下堆芯功率分布及功率峰因子等关键安全参数的实测值,在国内首次实现了功率台阶试验的“超短时等待”。应用结果表明,该技术将现行的平衡氙堆芯通量测量技术在每个功率平台的等待时间由传统技术超过24小时缩短至仅2小时^[109,110],在每台M310机组每次换料大修时可新增发电量约2000万千瓦时,创造出显著的经济效益。

3.2.2 单点校刻技术

压水堆机组堆外功率量程探测器(RPN)用于机组功率运行过程关键安全参数的在线监测、停堆保护等,核安全法规要求必须定期对RPN进行校刻,以保障机组的安全性。传统多点校刻技术通过快速、深度插入控制棒引发堆芯工况扰动后再实施校刻试验,极大地影响了机组的安全性。单点校刻技术依赖Bamboo-C软件的高精度和可靠性,仅需实施单次堆芯通量测量试验,通过变工况条件下堆外探测器响应的高精度数值模拟,代替现行的多点校刻技术依赖的多次(一般为5~6次)堆芯工况扰动试验,不仅进一步提升了机组运行安全性,而且显著缩短了每次校刻试验所需时间。应用结果表明,该技术对“华龙一号”、M310机组RPN校刻系数对轴向功率偏差和功率水平等关键指标参数的误差均值分别为0.24%(限值要求3%)和0.30%(限值要求5%),满足工程应用的精度和可靠性需求^[111,112]。

3.2.3 改进动态刻棒技术

压水堆机组在每次启动达到零功率临界状态后,核安全法规要求必须开展控制棒价值刻度试验,以保障机组安全运行所需停堆裕量的要求。传统动态刻棒技术通过对连续插棒过程RPN电流首先进行静态空间因子修正、然后进行动态空间因子修正等复杂过程实现控制棒价值刻度,存在流程复杂、人因依赖大等问题。改进动态刻棒技术通过连续插棒对RPN电流影响机理分析,创新性地定义瞬态空间因子综合考虑控制棒连续下插对中子通量密度的空间影响,代替现行的动态刻棒技术需要同时依赖静态空间因子和动态空间因子分步、联合修正的做法,极大地简化了控制棒价值刻度的数据处理流程。工程验证结果表明,该技术对

“华龙一号”、M310、VVER1000等机组多个燃料循环控制棒价值刻度误差均满足限值要求^[113,114], 具备工业应用所需的精度和可靠性。

4 总结与展望

NECP团队基于20余年来在核反应堆高保真计算中子学方法方面的基础研究, 自主研发了一系列基于先进方法的核反应堆中子学分析核心软件, 并成功应用于“华龙一号”、“国和一号”、EPR、AP1000等先进压水堆, 中国实验快堆、中国示范快堆等钠冷快堆, 高温气冷堆, 铅铋堆等小型反应堆, 以及聚变堆包层等多

个国家重大工程项目中, 在我国先进核反应堆的设计研发和在运核电厂的运行维护中发挥了重要作用。

NECP团队将面向我国核能与核动力对于先进核反应堆的重大需求, 不断完善计算方法, 充分应用人工智能、大数据、数字孪生、高性能计算等方面的最新技术, 提升软件产品的性能和功能, 同时加强与软件用户的联动, 提升软件的用户体验, 打造软件的用户生态, 在超高通量实验堆、多用途小堆、同位素生产堆、一体化快堆、钍基熔盐堆等更广泛领域内推广应用, 在堆芯设计、安全分析、技术支持、故障诊断、换料设计等环节发挥更大的作用。

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Summary for “先进核反应堆高保真中子学计算方法与应用研究进展”

Research advances in high-fidelity neutronics computational methods and applications for advanced nuclear reactors

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Nuclear reactor technology is of strategic importance to national energy security and defense capabilities. The design and safety analysis of reactor cores rely heavily on neutronics analysis. With the continuous advancement of reactor technology, traditional neutronics calculation methods based on various approximations are increasingly inadequate to meet the precision requirements of advanced reactor analysis. For over two decades, the Nuclear Engineering Computational Physics Laboratory (NECP Lab.) at Xi'an Jiaotong University has conducted systematic research and development on high-fidelity computational methods characterized by “high-precision, high-resolution, and high-confidence”, aiming to address the needs of advanced nuclear reactor development.

This paper comprehensively reviews the NECP Lab's progresses in key theoretical methods for high-fidelity neutronics analysis.

NECP lab has developed advanced methods for producing reactor-specific nuclear cross-section libraries. This includes novel approaches for evaluating thermal neutron scattering law data for crystalline and liquid materials using first-principles calculations, overcoming limitations of traditional evaluations. Furthermore, methods for nuclear data uncertainty analysis, based on sampling techniques, and data assimilation, utilizing generalized least squares and Bayesian inference, have been established to quantify and reduce computational biases.

Significant innovations have been made in resonance self-shielding calculations, which are critical for accurately treating the sharp variations in neutron cross-sections of nuclides like uranium and plutonium. The team has developed a comprehensive resonance calculation system applicable to various reactor spectra. Specific methods address challenges in different reactor types: for commercial pressurized water reactors (PWRs), global-local coupling methods and equivalent geometry techniques enhance efficiency and geometric adaptability; for dispersion particle-fueled reactors, improved global-local methods can handle double heterogeneity; and for fast reactors, methods like TONE and point-wise/ultra-fine-group approaches effectively manage the hard spectrum and multiple resonant nuclides.

For solving the neutron transport equation, NECP Lab has advanced both deterministic and stochastic (Monte Carlo) numerical methods. Developments on deterministic methods include stabilized finite element methods and transmission probability methods on triangular meshes for complex geometries, as well as enhanced method of characteristics (MOC) schemes with improved ray-tracing, CPU-GPU heterogeneous parallelism, and 2D/1D coupling for 3D simulations. For Monte Carlo methods, techniques like consistent adjoint driven importance sampling (CADIS) and its variants, along with adaptive weight window generation, drastically improve the computational efficiency for deep-penetration radiation shielding problems.

Based on these theoretical advancements, the NECP Lab has independently developed a full-fledged suite of high-fidelity neutronics analysis software with complete intellectual property rights. Key software packages include: NECP-Atlas: for processing evaluated nuclear data files (including the modern GNDS format) into application-ready libraries. NECP-Bamboo Series: for PWR core physics analysis, including the industrially validated Bamboo-C and the high-resolution pin-by-pin code NECP-Bamboo2.0. NECP-X: a high-fidelity deterministic numerical reactor code capable of full-core steady-state, transient, and depletion simulations with multi-physics coupling. NECP-SARAX: a fast reactor analysis code system already validated against international benchmarks and zero-power/reactor start-up tests. NECP-Panda: for the physics analysis of pebble-bed HTGRs, incorporating treatments for double heterogeneity. NECP-Hydra & NECP-MCX: for radiation shielding calculations using deterministic (discrete ordinates) and hybrid Monte-Carlo/deterministic methods, respectively, both recognized as fully independent and controllable industrial software.

These software systems have been extensively applied to support the design and operation of various advanced reactors in China. Applications include the nuclear design of special micro-reactors, ex-core neutron field analysis for large pool-type sodium-cooled fast reactors, and several operational support technologies for PWR nuclear power plants. The latter includes dynamic flux mapping, single-point calibration of ex-core detectors, and improved dynamic rod worth measurement techniques, which have enhanced both the safety and economic performance of operating reactors.

nuclear reactor physics, neutronics calculation, nuclear data processing, pressurized water reactor, fast reactor, high-temperature gas cooled reactor

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