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用MCNP程序计算核燃料废包壳缓发裂变中子形成的热中子通量密度

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摘要 用Monte Carlo方法计算核燃料废包壳缓发裂变中子形成的热中子通量密度分两步进行:第一步,计算出外中子源在包壳中生成的缓发裂变中子;第二步,计算这个缓发裂变中子源在探测器中所形成的热中子通量密度。为利用现有的MCNP程序进行计算,编制了有关的缓发裂变中子源生成及抽样子程序和体通量统计估计方法的记数子程序。计算表明:针对解决所遇到的深穿透问题,体通量统计估计法比径迹长度法要好些。

关键词 缓发裂变中子 Monte Carlo方法 通量密度

分类号

扩展功能

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COMPUTING WITH MCNP THE THERMAL FLUX DENSITY DUE TO THE DELAYED FISSION NEUTRON GENERATED IN PROCESSING THE CLADS OF BURNED-UP FUEL ROADS

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Abstract Describe a two-step procedure to calculate, by monte Carlo method, the thermal neutron flux den-sity produced by delayed neutron from fission in processing the clads of spent fuels: first, to calculatethe number of delayed fission nuetrons produced by external neutron source and its spatial distributionover the basket containing the clads of spent fuel; Second, to calculate the thermal neutron flux densitygenerated by this delayed neutron source. We have made some subroutines to complete the calculationwith existing MCNP, such as generating the spatial distribution of delayed neutrons, sampling from thisdistribution, and counting the volume flux of the detectors by the method of statistical estimate. Thecomputational results show that the method of statistical estimate of volume flux is better than themethod of track length in solving the deep penetration problem.

Key words Delayed fission neutron Monte Carlo method Flux density

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