NEER FY 2001
PROJECT ABSTRACT

Grant Number: 01ID14106

Project Title: Adaptive Neutron Transport Methods for Reactor Transient Analysis

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Abstract: The solution of the multigroup time-dependent Boltzmann transport equation with high order angular flux approximation can pose a prohibitive computational burden for practical reactor problems. Low order methods such as the two energy group diffusion equation are much more computationally efficient and have become the staple of reactor analysis within the nuclear industry. Several recent applications present a serious challenge to the adequacy of two group diffusion theory for LWR analysis. These applications would include some of the Generation IV advanced reactor designs, as well as some of the existing LWRs in which the core is more heterogeneous because of increases in the fuel discharge burnup or the irradiation of weapons grade MOX fuel.

The computational burden for higher order multigroup transport solutions in which each fuel pin is modeled explicitly can increase by at least two orders of magnitude. The work proposed here is to investigate methods that will allow higher order, multigroup transport methods to be used for practical reactor transient analysis. The innovation proposed is to employ adaptive methods for treating the spatial, angular, and time variables in the Boltzmann transport equation. Higher-order spatial and angular approximations will be invoked only in the regions of the core and at times during the transient where it is necessary in order to meet target accuracy. The methods developed here will be demonstrated within the framework of the US DOE code VARIANT and the US NRC code PARCS in order to insure the general applicability of the methods developed and to demonstrate their usefulness for practical reactor problems.