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Updating the Lambda modes of a nuclear power reactor

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ABSTRACT

Starting from a steady state configuration of a nuclear power reactor some situations arise in which the reactor configuration is perturbed. The Lambda modes are eigenfunctions associated with a given configuration of the reactor, which have successfully been used to describe unstable events in BWRs. To compute several eigenvalues and its corresponding eigenfunctions for a nuclear reactor is quite expensive from the computational point of view. Krylov subspace methods are efficient methods to compute the dominant Lambda modes associated with a given configuration of the reactor, but if the Lambda modes have to be computed for different perturbed configurations of the reactor more efficient methods can be used. In this paper, different methods for the updating Lambda modes problem will be proposed and compared by computing the dominant Lambda modes of different configurations associated with a Boron injection transient in a typical BWR reactor.

1. Introduction

An important problem in Nuclear Reactor Physics is the neutron transport equation [1]. This equation models the neutron flux on a given space composed by the position of the neutrons, their energy, and their velocity vector. A simplified version of this equation assumes that the energy of the neutrons is discretized into two energy groups and that the flux is isotropic, leading to the two energy group neutron diffusion equation [1]. A problem related with this equation is the generalized eigenvalue problem,

$$\mathcal{L}\Phi_i = \frac{1}{k_i}\mathcal{M}\Phi_i, \tag{1}$$

where

$$\mathcal{L} = \begin{bmatrix} -\vec{\nabla}\cdot(D_1\vec{\nabla}) + \Sigma_{a1} + \Sigma_{12} & 0 \\ -\Sigma_{12} & -\vec{\nabla}\cdot(D_2\vec{\nabla}) + \Sigma_{a2} \end{bmatrix},$$

is the neutron loss operator and

$$\mathcal{M} = \begin{bmatrix} \nu\Sigma_{f1} & \nu\Sigma_{f2} \\ 0 & 0 \end{bmatrix}, \qquad \Phi = \begin{bmatrix} \Phi_1 \\ \Phi_2 \end{bmatrix}$$

is the neutron production operator and the neutron flux. In the above equation the macroscopic cross sections for the first energy group are D_1 (the diffusion coefficient), Σ_{a1} (the absorption cross section), Σ_{12} (the scattering cross section from group 1 to group 2), $\nu\Sigma_{f1}$ (the average number of neutrons produced in each fission multiplied by the fission cross section),

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