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Three dimensional perturbation analysis of a fast reactor core

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1. Introduction

Perturbation analysis is usually performed for fast reactor cores to estimate the reactivity changes due to the changes in core properties like core dimensions, material concentrations and temperatures. These can be also used as an alternative method for estimating safety parameters like sodium void worth, Doppler coefficient etc. Compared to conventional direct method of finding reactivity changes, it is having the advantage of less convergence errors for small perturbations. Additionally, as an effective and simple method for finding the spatial distribution of reactivity change, these results can be used for transient analysis of fast reactor cores.

Safety analysis of various fast reactor cores that are already designed or under development in India has been carried out with the results obtained with first order perturbation theory. Safety analysis of Prototype Fast Breeder Reactor (PFBR), a 500 MWe fast reactor to be commissioned soon in Kalpakkam, India (Chetal et al., 2006), has been carried out with the help of perturbation code system ALCIA-LMI-NEWPERT (John, 1984). NEWPERT is a 2D perturbation code based on 1st order perturbation theory which uses the normal and adjoint flux of the reference core, obtained from the 2D diffusion code ALCIALMI. Such first order perturbation worths have mainly been used to investigate the time dependence of core reactivity during ULOFA (unprotected loss of flow accident) and UTOPA (unprotected transient over power accident), before the core transforms to the dis-assembly phase. The transient behavior of MOX fuelled PFBR core during a ULOFA before it enters in the dis-assembly phase is studied and reported (Harish et al., 2009).

Later, the 2D perturbation code NEWPERT is modified by adding the option of exact perturbation calculations and named PERTX. Then, the first order and exact perturbation analysis of a typical FBR core has been performed with the modified code and the results are compared with those obtained with ERANOS (Riyas et al., 2013). Later the exact perturbation results have been used to test the adequacy of first order results for the transient analysis of a MOX fuelled fast reactor core. The study reveals that, up to the start of dis-assembly phase, the material displacements are less such that the first order methods are well sufficient to simulate the core reactivity. The same scheme of perturbation analysis has been performed for the safety and transient analysis of metallic fuelled cores also; the relative safety performance of metallic fuelled cores during transients has been studied as a function of core size and reported (Sathiyasheela et al., 2011; Riyas and Mohanakrishnan, 2014).

Presently, efforts have been made to develop a three dimensional perturbation code (PERT3D) so that it can be used along with 3D diffusion code FARCOB, which is extensively used for the static core physics and burn-up calculations of PFBR core (Mohanakrishnan, 2008). Main steps towards the development of code PERT3D, the results of the comparative study with the existing 2D code and with the 3D perturbation modules available with ERANOS are briefly outlined in this paper.

2. Scope of the present analysis

In the 2D perturbation analysis, the perturbation worths are obtained in two dimensional planes which is an approximation to the actual 3D hexagonal core. When it is approximated with 2D R-Z geometry, symmetry along the azimuthal directions in the core is assumed. Actual radial zones of the core regions are approximated by annular rings in RZ model. At the same time, axial details of the core can be retained in 2D calculations too. The disadvantage in this approach is that, the reactivity change during a perturbation for a particular subassembly (SA) cannot be obtained (except for central SA). In addition, the worths of half rings also cannot be estimated accurately.

The development of 3D perturbation code will solve these issues of approximations taken in the 2D analysis with respect to the core geometry. Using the normal and adjoint fluxes generated by FARCOB, the new code will provide distribution of total reactivity change as the contributions from different SAs which constitute the core.

Additional advantages can be pointed out in terms of the reactivity change and its implications on transient analysis. For example, sodium void worth experiments are usually proposed for fast reactor cores cooled by sodium to validate the codes and nuclear data used for the core physics design. At present, more sophisticated methods like MCNP are used for such predictions. The proposed 3D perturbation methods can also be easily attempted for such analysis, though the method is approximate with respect to the complexities involved in the test SA design used in the experiment. Similarly, the reactivity change due to gas entry problems (Debanwita et al., 2017) in the reactor core can also predicted with the newly developed 3D perturbation code even with an

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asymmetric distribution of entrant gas inside core. Another main area of safety concern is with the reactivity changes during a seismic event. Such event may leads to SA bowing, oscillations and associated change in reactivity. A possible core deformation by some other causes can also leads to reactivity changes (Shikhov, 1960). During such scenarios, the 3D perturbation code can provide better predictions of reactivity changes as a function of core radial and axial displacements. Even though, the new proposed code is capable now for the conventional perturbation analysis for safety analysis, the above mentioned additional features of perturbation method can be accomplished through proper modifications of the code and input. In general, the approach is essentially important as this gives more realistic reactivity changes due to perturbations in the core and it will help to improve the existing transient analysis tools.

3. Formulation of the code

As mentioned in the previous section, the 3D diffusion code FARCOB provides the normal and adjoint fluxes for calculation. FARCOB treats the fast reactor core in hexagonal geometry by dividing a hexagon representing a sub-assembly (SA) in X-Y plane into 6 equilateral triangles, the side of which is decided by the SA pitch. It allows variable mesh size along the Z direction. The physical geometry of the reactor core can be seen as a collection of triangular prisms of different heights and filled with different materials/mixtures.

The fundamental eigen-value problem of a neutron multiplying system in the reference and perturbed cases can be represented as (Ronen, 1986);

$$L\Phi(x) - \lambda P\Phi(x) = 0 \tag{1a}$$

$$\text{and} \quad L'\Phi'^{(x)} - \lambda' P'\Phi'(x) = 0 \tag{1b}$$

where L, P, λ and Φ are (loss and production) operators, the eigen-value and flux of the reference system, similarly L′, P′, λ′ and Φ′ are the operators, eigen-value and flux for the perturbed system. The exact variation of reactivity change, Δρ due to the perturbation in the operators L and P can be represented by (Ronen, 1986; Ruggieri et al., 2006),

$$\Delta\rho = -\Delta\lambda = \left(1-\frac{1}{k}\right) - \left(1-\frac{1}{k'}\right) = \frac{\Delta k}{kk'} = \frac{\langle \Phi^{+}(\lambda'\Delta P - \Delta L)\Phi' \rangle}{\langle \Phi^{+} P \Phi' \rangle} \tag{2a}$$

where Φ^{+} is the adjoint flux of the unperturbed core. k and k' are the neutron multiplication factors of un-perturbed and perturbed systems and they are the inverse of the respective eigen-values.x is the symbolic representation of the independent variables of energy and space in the diffusion approximation.

ΔL, ΔP are the variations in the loss and production operators respectively.

For first order approximations, both the perturbed and unperturbed fluxes are assumed to be same and the reactivity change will be,

$$\Delta\rho = -\Delta\lambda = \frac{\Delta k}{k^2} = \frac{\langle \Phi^{+}(\lambda\Delta P - \Delta L)\Phi \rangle}{\langle \Phi^{+} P \Phi \rangle} \tag{2b}$$

The detailed expressions used in the calculations of both exact and first order reactivity changes due to a perturbation can be seen from the reference given (Riyas et al., 2013).

For calculating the neutron leakage contribution to the total loss term in the given expressions (Eqs. (2a) and (2b)), it is essential to find the change in diffusion coefficients due to perturbation in a triangular lattice, as FARCOB uses centre mesh differencing scheme with triangular meshes in the *X–Y* plane (Mohanakrishnan, 2008). Each triangular mesh point in the hexagonal SA represent a triangular prism and have five 5 neighbors, 3 in the radial and 2 in the axial directions as seen in Fig. 3.1. For mesh '*i*', let '*j*' represent its three neighboring meshes in radial directions and let '*k*' denote the two neighboring meshes in *Z* directions. The diffusion coefficient in the radial and axial directions has been defined separately in the code FARCOB to incorporate the radial and axial neutron leakages. Diffusion coefficient is defined in the radial directions for the i^{th} triangular mesh point with its three nearest radial neighbors as,

$$D_{ij} = \frac{2}{3(\Sigma_i + \Sigma_j)} \quad j = 1, 2, 3 \tag{3}$$

The 'Σ' represents the transport cross-sections, Σ_i is the transport cross-section for homogeneous mixture filled in the triangular prism which is represented by the ith mesh point. The diffusion coefficient for the mesh point 'i' in the axial directions is defined with the nearest axial neighbors above and below as,

$$D_{ik} = \frac{d_i + d_k}{3(d_i\Sigma_i + d_k\Sigma_k)} \quad k = 1, 2 \tag{4}$$

In contrast to radial mesh widths, axial mesh size can be varied (Mohanakrishnan, 2008). Let the region of interest is perturbed, the diffusion coefficient before and after perturbation is represented by D_{ij} and D'_{ij} respectively. Then, the change in diffusion coefficient in the radial directions can be obtained as;

$$\text{Since,} \quad D_{ij} = \frac{2}{3(\Sigma_i + \Sigma_j)} \quad \text{and} \quad D'_{ij} = \frac{2}{3(\Sigma'_i + \Sigma'_j)} \tag{5}$$

The change in diffusion coefficient is given by,

$$\delta D_{ij} = D_{ij} - D'_{ij} = \frac{2}{3} \times \frac{(\Sigma'_i + \Sigma'_j) - (\Sigma_i + \Sigma_j)}{(\Sigma_i + \Sigma_j)(\Sigma'_i + \Sigma'_j)} \tag{6a}$$

$$\text{or,} \quad \delta D_{ij} = \frac{3}{2} \times D_{ij} \times D'_{ij} \times ((\Sigma_i - \Sigma'_i) + (\Sigma_j - \Sigma'_j)) \tag{6b}$$

$$\textit{Thus,} \quad \delta D_{ij} = \frac{3 \times D_{ij} \times D'_{ij}}{2}(\delta\Sigma_i + \delta\Sigma_j) \quad j = 1, 2, 3 \tag{7}$$

There, we have used Eq. (5). In the similar fashion, for axial directions, the corresponding change in diffusion coefficient between two mesh points is given by,

$$\delta D_{ik} = \frac{3 \times D_{ik} \times D'_{ik}}{d}(d_i\delta\Sigma_i + d_k\delta\Sigma_k) \; ; \quad d = d_i + d_k \quad k = 1, 2 \tag{8}$$

Where D and D′ are the diffusion coefficients for the reference and perturbed core defined for the pair of mesh points. The change in diffusion coefficients then can be used to find the change in radial and axial leakages caused by perturbation, and then to get the leakage contribution to the net reactivity change.

As mentioned before for first order approximations, the perturbed flux in Eq. (2) will be replaced by the unperturbed reference flux assuming the perturbation is small. In addition, for first order approximation, the diffusion coefficients D = D′ in Eqs. (7) and (8).

4. Reference core used in the analysis

A 500 MWe fast reactor (FR) core is used as a reference core for this analysis. The radial core configuration is shown in Fig. 4.1. The radial representation of SA in the figure and in the diffusion calculations is limited up to 13 rings including central ring. The core has two fuel enrichment zones of core-1 and core-2 and surrounded by two rows of blanket SA along the radial directions. An active fuel column of 100 cm is provided in the core and is surrounded by lower axial blanket (LAB) and upper axial blanket UAB) of thickness 30 cm each. Reactor control and shutdown are accomplished by two types of control rods, Control and Safety Roads (CSR) and Diverse Safety Roads (DSR). Important core design parameters of the reference core are listed in Table 4.1 (Riyas et al., 2013).

5. Scheme of calculations used in the 3D perturbation analysis

A schematic of the calculation procedure used in the 3D