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Method for accounting for macroscopic heterogeneities in reactor material balance generation in fuel cycle simulations

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HIGHLIGHTS

- Describes addition of spatially dependent power sharing to a previous methodology.
- The methodology is used for calculating the input and output isotopics and burnup.
- Generalizes to simulate reactors with strong spatial and flux heterogeneities.
- Presents cases where the old approach would not have been sufficient.

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ABSTRACT

This paper describes the addition of spatially dependent power sharing to a methodology used for calculating the input and output isotopics and burnup of nuclear reactors within a nuclear fuel cycle simulator. Neutron balance and depletion calculations are carried out using pre-calculated fluence-based libraries. These libraries track the transmutation and neutron economy evolution of unit masses of nuclides available in input fuel. The work presented in the paper generalizes the method to simulate reactors that contain more than one type of fuel as well as strong spatial and flux heterogeneities, for instance breeders with a driver–blanket configuration. To achieve this, spatial flux calculations are used to determine the fluence-dependent relative average fluxes inside macroscopic spatial regions. These fluxes are then used to determine the average power of macroscopic spatial regions as well as to more accurately calculate region-specific transmutation rates. The paper presents several cases where the fluence based approach alone would not have been sufficient to determine results.

1. Introduction

The fluence based neutron balance method, used by the nuclear reactor simulator Bright-lite (Flanagan et al., 2014), calculates input and output compositions of fuel in nuclear fuel cycle simulators. The method to be used to calculate the blending ratio of multiple streams of arbitrary isotopic composition in order to obtain input fuel compositions that meet specified burnup, fluence, or conversion ratio objectives.

Bright-lite operates by characterizing the evolution of the neutron balance for each nuclide present in the fuel on the basis of fluence. The method uses low-order approximations to account for the effects of micro heterogeneities in the fuel lattice on the neutron balance, for instance flux differences between fuel and coolant regions. Effects of larger heterogeneities in the core, such as variations between batches having resided in the core for different lengths of time, are calculated by assuming that each macroscopic region is maintained at equal fission power density. This equal power sharing assumption was shown to lead to adequate results for a range of reactors and is accurate in an approximate sense as fuel managers strive to adjust core loadings so that power is distributed as uniformly as possible.

The methodology defined in this paper uses macro-region based coupled spatial flux and burnup calculations to enable the fluence based approach to treat strongly heterogeneous cores and fuel loadings. Homogeneous regions are defined within the reactor core at both the micro (fuel-cladding regions) and macro (batches) levels. Each region has a unique, time-dependent relative flux, which is used to advance the fluence of the material(s) within that region.

Eliminating the equal power sharing assumption enables potentially better treatment of batch-level homogeneous cores and grants a fuel cycle simulator the ability to dynamically model heterogeneous cores with reloads whose input and output isotopics evolve over time. For example, a homogeneous core cannot be

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used to model a breeder reactor due to no-power or low-power producing regions such as breeding blankets. In addition, the heterogeneous approach will allow irradiation targets, for instance americium-loaded transmutation pins, to be modeled. The new method will also improve results for mixed fuel types (such as uranium and plutonium in light water reactors) which had previously been treated as spatially decoupled. It is important to note that this method still requires its users to generate cross sections which are appropriate to the neutron spectra within the fuels or targets.

The next section will review methods used by nuclear fuel cycle (NFC) simulators to account for evolving input isotopics when determining input fuel composition and output burnup and composition. Subsequently, the elementary diffusion-theory based methodology which replaces the zero-dimensional relative flux calculation of Bright-lite is described. Results using the new methodology are presented next, followed by conclusions.

2. Input composition calculations

NFC simulators have a method for specifying the output composition and burnup if the composition of the input fuel is specified. This section reviews two methods to determine an input composition when multiple material streams are available to be blended. The accuracy of these methods is dependent on the number of nuclides tracked as well as the available one- or multi-group cross section data. Their general goal is to determine the input fuel composition that meets user- or fuel cycle simulator-defined output composition, reactivity, or burnup targets.

2.1.1. The reactivity worth approach

This method, which was proposed for implementation in VISION (Yee, 2008), is used to match the reactivity of the available (usually recycled) fuel to that of a reference fuel composition specified for the reactor. The available fuel streams are mixed so that the reactivity of the fresh resultant mixture represents a minimized change from the reference.

In order to determine the reactivity of the available fuel, first every nuclide in its composition is categorized according to the blending stream to which it belongs. For instance, if a uranium and a transuranic stream are being blended, nuclides of the U and TRU type are considered. Next, these nuclides' fractions in the reference fuel mixture are determined within their category and used with available cross section data to calculate the terms in the six-factor formula. A reactivity worth is then calculated for each nuclide. The reactivity worths are used to adjust the mass ratio of the streams being blended, leading to the fresh fuel that provides the best initial reactivity match to the reference. This method does not directly consider differences in the evolution of the multiplication factor with burnup as the isotopic content of the fuel is changed, so it would likely be most appropriate if the perturbations from the reference cases remain small. Due to this limitation, the Bright-lite methodology does not base its calculations on initial reactivity.

2.1.2. The D-factor method

This method, which has been utilized by COSI (Boucher and Grouiller, 2005), focuses on the net neutron contribution of each initial nuclide present in the fuel as it transmutes into fission products. First, the neutron consumption per fission of each nuclide i (D_i) is defined as the net "number of neutrons needed to transform the nucleus and its reaction products into fission products" (OECD, 2006). Therefore a positive D corresponds to a net neutron consumption by the species and its descendants, and a negative D to production of neutrons. The D-factor captures the characteristics of each reactor via cross sections which depend on the flux spectrum of the reactor and the burnup and residence time of fuel. The relative fraction of each nuclide (m_i) in fresh fuel is used to find the D-factor for the fuel (D_{FUEL}) using Eq. (1).

$$D_{FUEL} = \sum D_i m_i \tag{1}$$

Then, Eq. (2) is used to determine core neutron balance.

$$G = S_{ext} - D_{FUEL} - losses \tag{2}$$

where S_{ext} is a potential external neutron source and *losses* is the neutron losses (by leakage) per fission in the reactor. When G, the neutron surplus, is positive, the core is assumed to maintain criticality during burnup. The objective of the methodology is to create valid input fuel compositions for use in COSI by blending available fuel sources to obtain $G = 0$.

3. Methodology

The fluence based neutron balance method used in Bright-lite can simulate multiple batches with unique burnup histories by tracking key fluence-dependent properties of each batch. The method pre-calculates and parameterizes reactor type-specific burnup and transmutation libraries using one energy group cross sections for every nuclide which can be present in fresh fuel. The cross sections are precomputed via coupled transport-burnup calculations using representative fuel configurations, compositions, and discharge burnups. Libraries are generated using an existing depletion calculation tool (such as ORIGEN (Bell, 1973)) for a unit mass of the nuclide using a characteristic flux for the fuel and reactor type. Each library contains a fluence-dependent neutron production rate, neutron destruction rate, burnup, and transmutation matrix for the nuclide.

Fig. 1 shows two example libraries for a representative light water reactor. Both plots illustrate the evolution of the contribution of an initial unit mass of a fuel isotope to the neutron balance. All transmutation and decay daughters of the initial masses are tracked, so their contributions are also reflected in the figures. The left follows an initial unit mass of U-235 where the neutron production rate remains greater than the destruction rate until the fluence becomes quite large and most of the U-235 atoms initially present have become fission or transmutation daughters. On the other hand, in the right figure for U-238, the neutron destruction rate exceeds the production rate for all fluences, although both increase with fluence due primarily to the buildup of Pu-239. During runtime these isotope libraries are combined for each macroscopic spatial region on a mass-weighted basis to generate a characteristic library of the desired fuel composition.

The methodology for discharge fluence calculation can be demonstrated using an example 3-batch reactor operating in steady state with known fuel compositions. Fig. 2 depicts the increase in batch fluences marching forward in time within one reactor cycle. Initially the first batch has zero fluence ($F^{(1)}(t=0)=0$), and the third batch has the highest.

To march forward through the cycle, a time step Δt (during which the magnitudes of the average fluxes in the region occupied by each batch are assumed constant) is implemented to calculate $\Delta F^{(n)}(t)$ for each batch. In general, if the volume and time averaged flux ($\Phi^{(n)}(t)$) in the fuel region of batch n over time step starting at t is known, the fluence increment $\Delta F^{(n)}(t)$ can be calculated for each batch:

$$\Delta F^{(n)}(t)\left[\frac{n}{\text{cm}^2}\right] = \Delta t[\text{s}] \cdot \Phi^{(n)}(t)\left[\frac{n}{\text{cm}^2}\cdot \text{s}\right] \tag{3}$$

The eigenvalue is calculated using the neutron production rates ($P^{(n)}(t)$), neutron destruction rates ($D^{(n)}(t)$), time averaged fluxes