# Simulation and Modelling of Uranium Burnup Using Neutron Reaction Networks and ODEs

Abhinand Satish

Project Overview:

This project details the development and implementation of a numerical simulation to model the burnup of Uranium chain-based fuel and its subsequent transmutation chain. How this would happen is by leveraging data from the neutron flux distributions, reaction cross-sections, and decay constant, a system of coupled first-order differential equations derived from Stacey’s “Nuclear Reactor Physics” book. The final model of this project will allow the user to input a file where a list of isotopes and their subsequent concentrations and neutron flux which would compute the evolution of the isotopic concentrations over time. With that in mind here is a list of requirements for the simulations to run:

* All 19 differential equations from Stacey’s textbook
* Neutron cross sections for each reaction type for the isotopes that are involved
* Materials of Fuel and their concentrations
* Using an input flux to calculation the reaction rate of each reaction

The above is basically a layout for the process that I used to calculate the isotopes produced and the specific outputs will be explained. Keep in mind that the goal of this project is to be able to show what the isotopes are and their concentrations in a particular time frame. This is a very common requirement as Nuclear Reactions have burnup and cool down times, so it is important to understand what your fuel looks like.

Stacey’s Ordinary Differential equations

In chapter 6.1 – changes in Fuel composition, page 199, we can see the Fuel Depletion-Transmutation-Decay Equations. The concentrations of the various fuel isotopes in a reactor are described by a coupled set of production-destruction equations. In the following equations three is going to be a two-digit superscript to identify isotopes in which the first digit is the last digit in the atomic number and the second digit is the last digit in the atomic mass. We represent the neutron reaction rate by , although the actual calculation may involve a sum over energy groups of such terms. Here is the list of isotopic concentrations are described by:

These 19 differential equations are 19 isotopes that are found in a Uranium Decay chain. Here is a list of all the isotopes and the codes to them so it is easy to correspond each equation to each isotope.

|  |  |
| --- | --- |
| **Isotope** | **Code** |
| U234 | 24 |
| U235 | 25 |
| U236 | 26 |
| U237 | 27 |
| U238 | 28 |
| U239 | 29 |
| Np236 | 36 |
| Np237 | 37 |
| Np238 | 38 |
| Np239 | 39 |
| Pu238 | 48 |
| Pu239 | 49 |
| Pu240 | 40 |
| Pu241 | 41 |
| Pu242 | 42 |
| Pu243 | 43 |
| Am241 | 51 |
| Am242 | 52 |
| Am243 | 53 |

So how we interact with these differential equations is by building a function that can include production and loss terms based on the isotope. For instance, we have an input isotope and then build production and loss terms based on that isotope from data stored. This would be easier than having to solve each differential equation every time rather than building the equations on a case-by-case basis.

Cross section and Flux implementation

So, the main reactions that lead to productions and loss terms are absorption, capture and n2n reaction. Based on these reactions we now need to get the cross-section data for each isotope for each reaction. This cross-section data will then need to be split into 175 bands as we need a finite number to be able to calculate the reaction rate with the flux. The reason why we do this is because we can have an infinite cross section grouped data and we need to be able to consolidate it into a finite number to calculate.