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Main author(s): B. Chmielarz, W. Gudowski (KTH)

Contributing author(s): T. Bäck (KTH), A. Tkaczyk (TARTU)

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Master's Thesis No. 1 offered by the project:

FANCSEE - Development of a fuel cycle code with Graphical User Interface

Błażej Chmielarz

Thesis supervisor: Wacław Gudowski, waclaw@kth.se

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Kungliga Tekniska högskolan – Nuclear Reactor Physics, Stockholm, Sweden

Executive summary

The purpose of the project was to assess and improve a nuclear fuel cycle code FANCSEE written by Torbjörn Bäck. The method of burnup and decay calculation, cross sections of PWR and BWR were based on ORIGEN. The original idea was to create a more accurate and user-friendly software than ORIGEN. The goals were supposed to be achieved by burnup calculation using smaller timesteps, accurate estimation of flux and implementation of a Graphical User Interface (GUI) where a user could create and run any nuclear scenario using point-and-click method. The flux calculation was never successfully implemented.

The basic principle of operation of FANCSEE is based around creation, burnup and decay of nuclear fuel and it operates by passing an isotope vector, describing concentration of 1307 different isotopes, between facilities present in a nuclear cycle. In FANCSEE, a nuclear fuel cycle code consists of mines, enrichment facilities, fuel fabrication facilities, reactors and waste repositories. Burnup of fuel is simulated through multiplication of the isotope vector by a burnup matrix exponential solution for a specific reactor and timestep. The burnup matrix is a general solution to a set of Bateman equations extended by reaction rate terms which reflect cross section of neutron interactions and neutron flux. The solution to the exponential of the matrix is calculated by approximation near origin using the first two terms of Taylor expansion and longer timesteps are calculated using formula $e^{At} = \left(e^{\frac{At}{m}}\right)^m$ which allows to express longer timesteps using shorter ones.

The assessment of the code was done through comparison of transients of ^{235}U,^{238}U,^{239}Pu and ^{241}Am isotopes with burnup results generated by a modified Serpent example based on NEA PWR benchmark and simulation of ALMARAZ-1 reactor by EVOLCODE, which uses a similar calculation method. All three codes used the same power and enrichment, fuel cycle length, fuel mass, the same burnup of 40 MWd/kgU and all were based on cross sections from a PWR. The results of unmodified FANCSEE showed large deviations from the reference results; ^{235}U burned too quickly, ^{238}U burned too slowly, ^{239}Pu burned too quickly and was generated too slowly. ^{241}Am maintained a constant gap between EVOLCODE and FANCSEE throughout burnup The differences at the end of the cycle were between 0,5% and 18%. Adjustment to initial fuel composition by addition of ^{234}U yielded nearly exactly the same results, with changes being below 1% at any point. Adjustment of cross sections of ^{235}U, ^{238}U and ^{239}Pu using a weighted average of single-group microscopic cross sections from Serpent. The new cross sections were, in general, significantly lower. The change brought the output of FANCSEE simulation much closer to the reference results, particularly in the case of ^{239}Pu. The reaction rate appeared too low in the latter half, though. With the conclusions that the reaction rate in ^{238}U and ^{235}U are lowered due to accumulation of fission products and transuranic elements, the last featured simulation run implemented recalculated cross

sections of samarium isotopes, ^{240}Pu, ^{241}Pu and ^{241}Am. The results remained quantitatively the same for ^{235}U, ^{238}U and ^{239}Pu but became unreasonable for ^{241}Am. Calculation of expected average flux in the simulations yielded a result which was 47% of value used in FANCSEE, raising a question about quality of implementation of burnup matrix exponential solution.

The conclusions are that FANCSEE needs development and benchmarking. The development should prioritize acceptation of new cross section libraries, reaction rate adjustments using either multiple solutions to the exponential throughout burnup or arbitrary readjustment of timesteps and implementation of a new method to calculate the exponential solution.