

Sensitivity Analysis and Uncertainty Quantification of FFTF Cycle 8C Using the NEAMS Workbench

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INTRODUCTION

This paper summarizes the sensitivity analysis and uncertainty quantification performed on the global reactivity feedback coefficients of the Fast Flux Test Facility (FFTF) Cycle 8C core configuration. The Fast Flux Test Facility (FFTF) was a 400-MWt, sodium-cooled, low-pressure, high-temperature, fast-neutron flux, nuclear fission reactor plant designed for the irradiation testing of nuclear reactor fuels and materials for the development of liquid metal fast breeder reactors (LMFBRs). The main focus of this study is to quantify the uncertainties in different reactor neutronic feedback coefficients based on the Cycle 8C core configuration [1] that was run at 50% power level and 100% flow, and that included some Gas Expansion Module (GEM) as a means of passive reactivity control. FFTF was fueled with plutonium-uranium mixed oxide (MOX), and contained 32 inner driver fuels and 48 outer driver fuels, 9 B₄C control rods, 9 GEM positions, 3 material testing positions, and 98 radial reflectors that were made up of SS-316 and Inconel-600. Figure 1 represents the geometrical view of FFTF with zero-angle rotation about the z-axis.

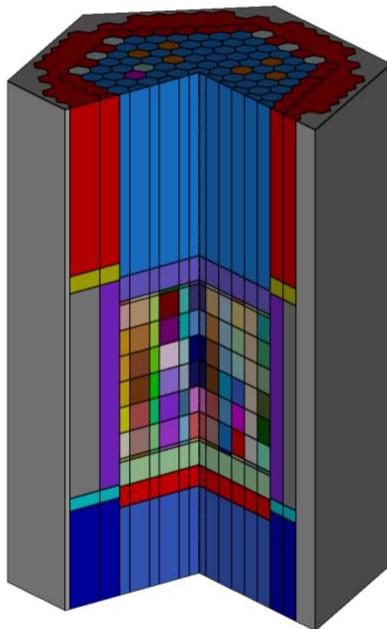


Fig. 1. Geometrical view of FFTF assemblies.

Benchmark analysis on this test performed during the FFTF experiments was carried out as an additional validation exercise of ANL sodium-cooled fast reactor neutronic and safety analysis capabilities. In particular, the impact from uncertainties in nuclear data was assessed.

METHODOLOGY

The FFTF core was modeled using the Argonne Reactor Computation (ARC) suite of deterministic neutronic codes for fast reactor analysis. It includes MC²-3 for multi-group cross-section processing, DIF3D for neutron and gamma transport calculation, and PERSENT for perturbation, sensitivity and uncertainty quantification calculations. The ARC codes have been integrated within the Nuclear Energy Advanced Modeling and Simulation (NEAMS) Workbench [2] through the python module PyARC [3].

The FFTF core model shown in Fig. 1 uses detailed description of each assembly, where each driver fuel segment is described with 8 axial regions. A total of 1210 unique regions are modeled with an assembly-wide radially homogenized composition.

The MC²-3 code [4] is used to calculate the 33-group cross-sections for the DIF3D/VARIANT code [5] employing the ENDF/B-VII.0 nuclear data library [6]. Reference homogeneous cross-section calculations employed three steps. The first step uses a fine 2082-energy-group structure and cross sections are condensed into 1041 energy groups. It is followed by a flux calculation step using the 2-D Sn transport solver TWODANT for an approximated equivalent 2-D cylindrical core with P3 scattering approximation and 1041 energy groups. During the third step, the cross sections are condensed into 33 groups using the flux spectrum obtained from TWODANT for each region. Heterogeneous cross-section treatment is applied in this second step to the driver fuel region using MC²-3 based on equivalent 1-D fuel pin model to account for the geometrical self-shielding [7].

The whole-core flux calculation was performed with the DIF3D code [5] using the variational nodal transport solver VARIANT with the 3rd-order angular flux and 3rd-order scattering approximations and the 33-energy group discretization. Gamma and neutron power per assembly area of the FFTF were calculated using GAMSOR code [8].

PERSENT [9] is a perturbation theory code for sensitivity analysis in three-dimensional geometry and transport theory. In terms of the reactivity feedback

coefficient calculations, first order perturbation theory was applied in order to calculate the following reactivity feedback coefficients:

- Fuel Doppler constants (flooded and voided),
- Fuel density,
- Structure density,
- Sodium density.

And direct perturbation calculation was used to calculate:

- Axial expansion,
- Radial expansion,
- Control and safety rods worth,
- Gas Expansion Module worth.

More information on the assumptions made in the calculations can be found in Ref. [1]. These coefficients are used to evaluate the inherent safety capability of the SFR based on transient simulations, which will be performed with the SAS4A/SASSYS-1 code [10]. The nuclear data uncertainty on neutronics feedback coefficients are estimated with the PERSENT code based on the COMMARA-2.0 covariance matrix [11]. The nuclear data uncertainties are therefore estimated on the eigenvalue, beta, lambda, and on the reactivity coefficients such as the fuel, structure, coolant density coefficients and the Doppler constant. The present paper summarizes the impact of different modeling approximations and of the nuclear data uncertainties, evaluated with the aforementioned neutronic codes integrated in the NEAMS Workbench [2, 3], on the FFTF Cycle 8C core model.

RESULTS

Table I illustrates the impact of different cross-section generation methodologies on the reactivity feedback coefficient results. The heterogeneous and homogeneous cross-section treatments and the one or two-step (with TWODANT) workflow within MC²-3 are compared. It is essential to note that no significant changes in the coefficients are observed in all the three cases that involve different cross-section generation methodologies with one-step (no TWODANT) or two-step (with TWODANT) workflow. Irrespective of the workflow, the heterogeneous cross-section treatment of the fuel yielded a very slight variation in coefficients in the case of all the reactivity coefficients when compared with homogenous cross-section treatment. The heterogeneous cross-section treatment and the two-step cross-section processing provide higher k-effective by about 350pcm and 300pcm respectively.

Turning to the nuclear data uncertainties, the eigenvalue is associated with an uncertainty of 0.79%, which is mainly due to the inelastic and ν reactions of Pu-239, and to the inelastic reaction of U-238. The Structure density, as indicated in Table II, contributes the largest positive reactivity feedback and its coefficient is associated with 17.74% uncertainty coming from nuclear data. This is followed by the Sodium density (13%), while the lowest

contribution was found in the Fuel density with 1.63%, respectively. The uncertainty coming from nuclear data should be propagated through the transient simulations in the future using the methodology developed in [12].

Table I: Comparison between reactivity feedback coefficients using different cross-section generation methodologies.

MC ² -3 heterogeneous treatment	False	False	True
MC ² -3 2-step procedure (with TWODANT)	True	False	True
k-eff	0.996488	0.993489	0.999957
Fuel dens. [$\Delta k/k$]	-4.57×10^{-3}	-4.61×10^{-3}	-4.54×10^{-3}
Structure dens. [$\Delta k/k$]	1.04×10^{-3}	1.05×10^{-3}	1.07×10^{-3}
Sodium dens. [$\Delta k/k$]	-1.25×10^{-4}	-1.28×10^{-4}	-1.26×10^{-4}
Doppler coeff. [$\Delta k/k$]	-4.36×10^{-3}	-4.26×10^{-3}	-4.28×10^{-3}
Radial exp. [$\Delta k/k$]	-6.31×10^{-3}	-6.35×10^{-3}	-6.28×10^{-3}
Axial exp. [$\Delta k/k$]	-3.25×10^{-3}	-3.27×10^{-3}	-3.23×10^{-3}

CONCLUSION

The FFTF Cycle 8C core was modeled using the NEAMS Workbench PyARC interface to compute the reactivity coefficients, the neutron and the gamma power distributions. These will be used for benchmark simulations using SAS4A/SASSYS-1. The present work discusses the impact of cross-section processing approximation and of nuclear data uncertainty on the eigenvalue and reactivity feedback coefficients. For FFTF, both the heterogeneous cross-section treatment and the TWODANT step were found to provide relatively marginal impact on the reactivity coefficients, but do impact the k-effective by 650pcm. In addition, the nuclear data uncertainty was propagated to the neutronic feedback coefficients and was found to be responsible for 1% to 18% uncertainty on some reactivity feedback coefficients. These uncertainties will be propagated through the LOFWOS transient in FFTF to see if the uncertainties coming from the nuclear data help explain some of the discrepancy observed between the best estimate results and the experiment.

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Table II: Uncertainty quantification results for major isotopes and reactions.

Coefficient		k_{eff}	Fuel dens.	Coolant dens.	Structure dens.	Flooded Doppler coeff.	Voided Doppler coeff.
		0.996488	-4.57E-03	-1.25E-04	-1.04E-03	-4.36E-03	-3.39E-03
Uncertainties [%]							
U238	nu	0.11%	0.07%	0.79%	1.74%	0.22%	0.24%
	Capture	0.19%	0.19%	2.50%	1.90%	0.80%	0.75%
	Elastic	0.21%	0.17%	2.96%	6.24%	0.61%	0.76%
	Inelastic	0.39%	0.50%	3.43%	5.79%	1.92%	2.12%
	Total	0.49%	0.51%	5.24%	8.91%	2.18%	2.39%
Pu-239	nu	0.07%	0.07%	0.76%	0.55%	0.18%	0.17%
	Fission	0.23%	0.33%	2.67%	2.12%	0.64%	0.67%
	Capture	0.29%	0.13%	0.94%	2.26%	0.85%	1.10%
	Inelastic	0.06%	0.18%	0.36%	0.73%	0.74%	0.87%
	chi	0.15%	0.06%	0.68%	1.74%	0.56%	0.58%
	Total	0.42%	0.40%	3.10%	3.73%	1.44%	1.68%
Pu-240	nu	0.11%	0.09%	0.87%	0.86%	0.23%	0.25%
	Fission	0.06%	0.08%	0.58%	0.51%	0.16%	0.17%
	Capture	0.07%	0.08%	0.69%	0.86%	0.36%	0.39%
	Inelastic	0.02%	0.03%	0.12%	0.14%	0.12%	0.13%
	chi	0.04%	0.00%	0.33%	0.53%	0.16%	0.18%
	Total	0.15%	0.15%	1.32%	1.45%	0.50%	0.54%
Na-23	Elastic	0.12%	0.61%	4.49%	8.08%	1.67%	0.79%
	Inelastic	0.04%	0.03%	3.02%	0.98%	0.39%	0.17%
	P1elastic	0.14%	0.40%	5.50%	5.77%	0.19%	0.10%
	Total	0.19%	0.73%	7.73%	9.98%	1.72%	0.81%
	Capture	0.10%	0.32%	1.18%	5.18%	0.51%	0.55%
Fe-56	Elastic	0.24%	0.94%	6.22%	3.68%	0.97%	1.13%
	Inelastic	0.09%	0.10%	0.87%	3.83%	0.44%	0.51%
	P1elastic	0.09%	0.27%	2.23%	0.26%	0.05%	0.08%
	Total	0.29%	1.04%	6.77%	7.37%	1.19%	1.36%
Total %		0.79%	1.63%	12.97%	17.74%	3.59%	3.66%