BLIND-PHASE RESULTS OF THE FFTF NEUTRONIC BENCHMARK

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Abstract

The International Atomic Energy Agency (IAEA) initiated a coordinated research project (CRP) in 2018 for the analysis of the Fast Flux Test Facility (FFTF) Loss of Flow Without Scram (LOFWOS) Test #13. This paper focuses on the neutronic results that were obtained during the blind phase of the benchmark. The comparison of the neutronic parameters focused on the eigenvalue, kinetics parameters, reactivity feedback coefficients, and assembly-wise power distribution. Eleven participating organizations from nine countries submitted their results using a wide variety of tools. Relatively good agreement is obtained between participants for the kinetics parameters (delayed neutron fraction and prompt neutron lifetime), Doppler coefficient, the safety and control rod worths, and on the neutron multiplication factor where the average estimate is close to 1.00, which is in good agreement with the experimental value for this critical reactor. The results of the Gas Expansion Modules (GEMs) worth display relatively good agreement for most participants, except for a few outliers. The agreement is less satisfactory for other reactivity parameters, such as the axial and radial thermal expansion coefficients, and on the density coefficients. Finally, satisfactory agreement is obtained on the power distribution in the fuel assemblies, while larger discrepancies are observed on the power estimates deposited in non-fissile assemblies.

1. INTRODUCTION

The International Atomic Energy Agency (IAEA) initiated a coordinated research project (CRP) in 2018 for the analysis of the Fast Flux Test Facility (FFTF) Loss of Flow Without Scram (LOFWOS) Test #13 [1, 2]. 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) [3].

With twenty-five organizations participating from thirteen countries, this CRP supports validation of sodium-cooled fast reactor neutronics and safety analysis tools and methods. The CRP began with a blind phase that ended in October 2020, during which each participant was tasked with performing their own independent model development and calculations for LOFWOS Test #13. Benchmark participants had the choice to use their own generated neutronic parameters for the transient analysis or to request the ANL-generated parameters.

In addition to the full transient benchmark, a neutronic benchmark was developed in order to compare methodologies used to generate neutronics parameters used in the transient simulations. Eleven organizations participated in this benchmark and provided results in the blind phase; those are summarized and discussed in the paper.

2. BENCHMARK DESCRIPTION

LOFWOS Test #13 was initiated at 50% power level and 100% flow and was conducted during the FFTF Cycle 8C core configuration [1]. The radial layout of the FFTF core cycle 8C is shown in Fig 1. FFTF was fueled with plutonium-uranium mixed oxide (MOX). During Cycle 8C, the core was loaded 32 inner driver fuel and 48 outer driver fuel (DF) assemblies, 6 B_4C control rods (CR), 3 B_4C safety rods (SR), 3 material testing positions (FMA, MOTA, ICSA), and 98 radial reflectors that were composed of SS-316 and Inconel-600 (REFL). This FFTF core configuration included 9 Gas Expansion Module (GEM) as a means of passive reactivity control. The core specifications include detailed structural material and depleted fuels compositions in every axial region of every assembly of this test reactor.

This benchmark provides code-to-code comparison of various neutronic parameters required for transients modeling:

- Neutron multiplication factor simulation, should be 1.0 since the core configuration modelled is in critical configuration.
- Delayed neutron fraction.
- Prompt neutron lifetime.
- Nominal power production for each assembly including fission and gamma heat.
- Total control and safety rods worth – obtained by changing the insertion depth (i.e. the bottom of the absorber) from the bottom of the lower insulator pellet up to the top of the upper insulator pellet.
- Global reactivity feedback coefficients:
    - Axial expansion coefficient [pcm/K] – 1% expansion calculated without moving control rods and assuming oxide fuel linear thermal expansion.
    - Radial expansion coefficient [pcm/K] –increasing the pitch of all assemblies by 1% while conserving the mass of all solid materials present in the core, assuming linear thermal expansion coefficient of SS-304.
    - Total GEM worth [pcm] – changing the sodium level within all the GEMs from the nominal level above the fuel region down to the inlet nozzle.
    - Fuel density coefficient [pcm/K] – reducing by 1% the density of all fuel isotopes, in fuel and insulator pellets, assuming volume thermal expansion of oxide fuel.
    - Structure density coefficient [pcm/K] – increasing by 5% the density of all structural components (SS-316 and Inconel 600), in all regions of the core, assuming volume thermal expansion of SS-316.
    - Coolant density coefficient [pcm/K] – reducing by 1% the density of sodium in all regions everywhere in the core, assuming volume thermal expansion of sodium.
    - Fuel Doppler Constant [pcm] – increasing the fuel temperature from 1000K to 2000K.

Eleven participating organizations from nine countries submitted their results for the blind phase of the benchmark. As shown in Table 1, participants used a wide variety of tools, with different nuclear data libraries (ENDF/B, ABBN, JENDL, JEFF), and different code solvers (stochastic and deterministic codes solving the diffusion or transport equations). The deterministic codes employed also rely on various multi-group cross-section processing methodologies. Such variety of methods used is extremely valuable, but even in the best-case scenarios where the models are fully consistent, it is expected that these differences in methods would cause noticeable discrepancies in the results obtained [4].

TABLE 1. FFTF NEUTRONIC BENCHMARK PARTICIPANTS

Country	Organization	Neutronics Code	Modeling Methods and codes	Cross-sections library (Energy group of transport solver)
China	**INEST**	NTC	Transport	HENDL
China	**NCEPU**	MGGC	Diffusion	ENDF/B
Germany	**HZDR**	Serpent-2	Monte-Carlo	ENDF/B.VII.1
Germany	**KIT**	ERANOS, PARTISN	Transport	KIT-72 (11)
India	**IGCAR**	FARCOB, MCNP4C	Diffusion, Monte Carlo	ABBN-93, ENDF/B-VIII.0
Italy	**Sapienza**	ERANOS, PHISICS	Transport	JEFF3.1.1
Japan	**JAEA**	MARBLE, MVP	Transport, Monte Carlo	JENDL-4.0
Russia	**IPPE**	-	-	ABBN-93(26)
Sweden	**KTH**	Serpent-2	Monte-Carlo	JEFF-3.2
Switzerland	**PSI**	Serpent-2	Monte Carlo	JEFF-3.1.1
U.S.A.	**ANL**	MC^2-3/DIF3D	Transport	ENDF/B.VII.0 (33)

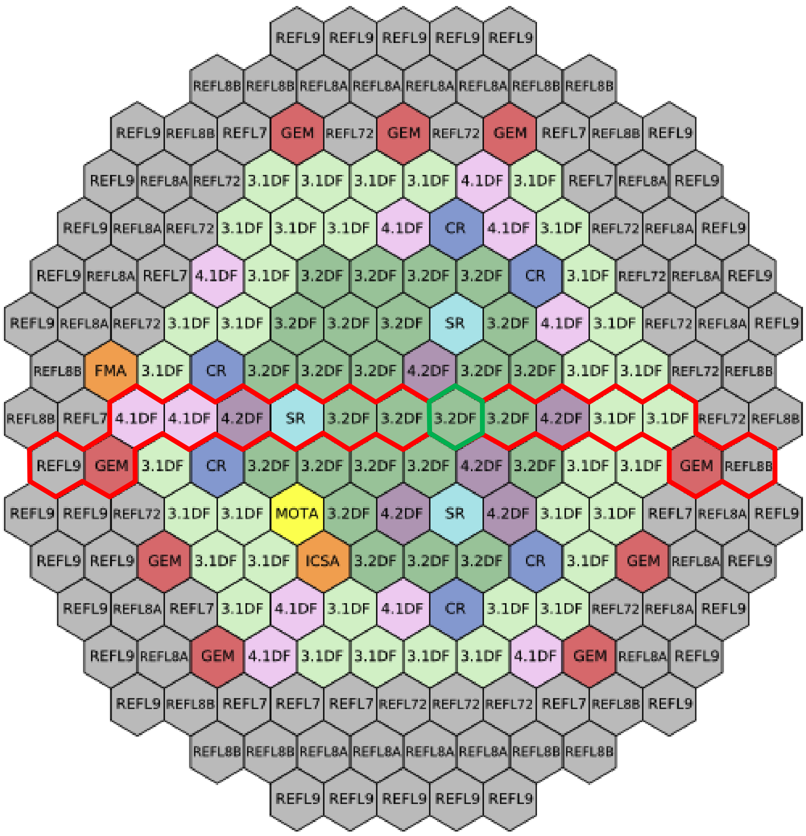

FIG. 1. Radial layout of FFTF core

3. RESULTS

The neutronics benchmark results submitted by the participants are summarized in Table 2. The results highlighted in yellow are obvious outlier values that were confirmed to be outside the average +/- 2-σ (after excluding them from the average and standard deviation calculations). Some participants omitted some results in their submission, as highlighted in green. The last column of the table displays the standard deviation divided by the average, which was computed after eliminating the highlighted outliers, to inform on the level of discrepancies observed in the results submitted. The figures presented in this section are a sampling of key parameter comparisons for the neutronics component of the blind phase of the CRP.

Variations in the radial power distribution in every FFTF assembly are shown in Fig. 2. Only seven participating organizations submitted assembly power information. Satisfactory agreement is obtained on the power distribution in the fuel assemblies (<11% standard deviation), while larger discrepancies are observed on the power estimates deposited in safety rods (~20%), GEMs (~40%), radial reflectors (15-40%), and in the MOTA and ICSA assemblies (~40%). Power provided by each participant along a line of assemblies across the core (highlighted in Fig .1) are displayed in Fig. 3. It can be observed that large power discrepancies are concentrated on the non-fissile regions, which is explained by the fact that some participants may have neglected gamma transport or used the diffusion approximation.

The results comparison in Table 2 shows that, after excluding a few outliers, relatively good agreement within 15% is obtained between participants for the neutron multiplication factor, kinetics parameters (delayed neutron fraction and prompt neutron lifetime), Doppler coefficient, fuel density coefficient, and the safety and control rod worths.

The comparison of the neutron multiplication factor submitted by participants is also illustrated by Fig. 4. The results obtained are all close to 1.00, which is in good agreement with the experimental value for this critical reactor. Variations in eigenvalue are typically less than 0.10%, and this level of discrepancy can be easily explained by the methods used by participants with different nuclear data libraries, stochastic versus deterministic codes, and multi-group cross-section processing methodologies [4].

These different methods and nuclear data can also explain the variation in the Doppler constant estimates shown in Fig. 5. However, some modeling inconsistencies are also likely to explain part of the discrepancy, especially for the outliers from groups using similar methods as other participants but calculating very different results. The results of the GEMs worth display relatively good agreement for most participants, as shown in Fig. 6, except the outlier groups, with one group predicting a positive feedback for the GEMs when the sodium level drops and two groups predicting significantly more negative values than the other groups.

The agreement is less satisfactory for other reactivity parameters, such as the axial and radial thermal expansion coefficients, but also for the structure, and sodium density coefficients, where the discrepancy on the results obtained exceeds 20%. This discrepancy is especially noticeable for the sodium and structure density coefficients where the results are widely spread out making it impossible to identify outliers. Those discrepancies are likely coming from inconsistent approaches in computing these coefficients. For instance, these density coefficients may not all have been computed over the same axial and radial regions, which should include the core, reflector and shielding regions, or may have been renormalized using the linear thermal expansion coefficient instead of the volumetric thermal expansion coefficient.

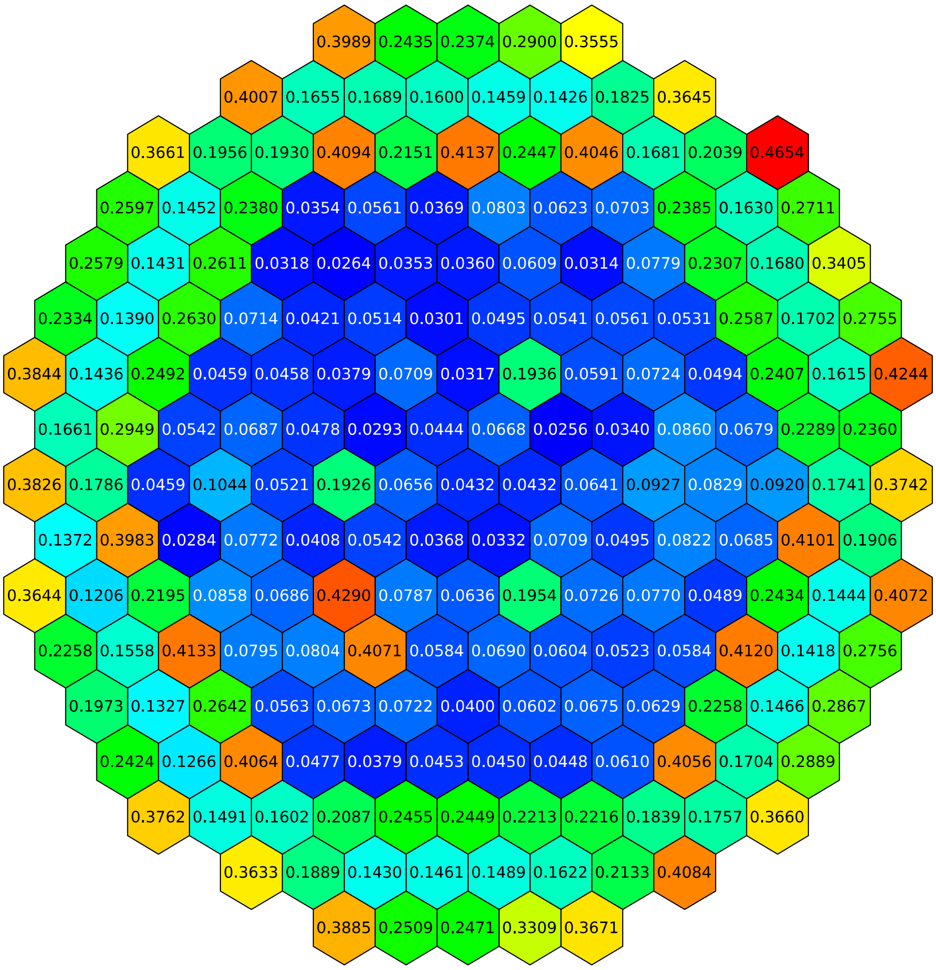

FIG. 2. Variation in Radial Power Profile Results (Standard Dev. / Average)

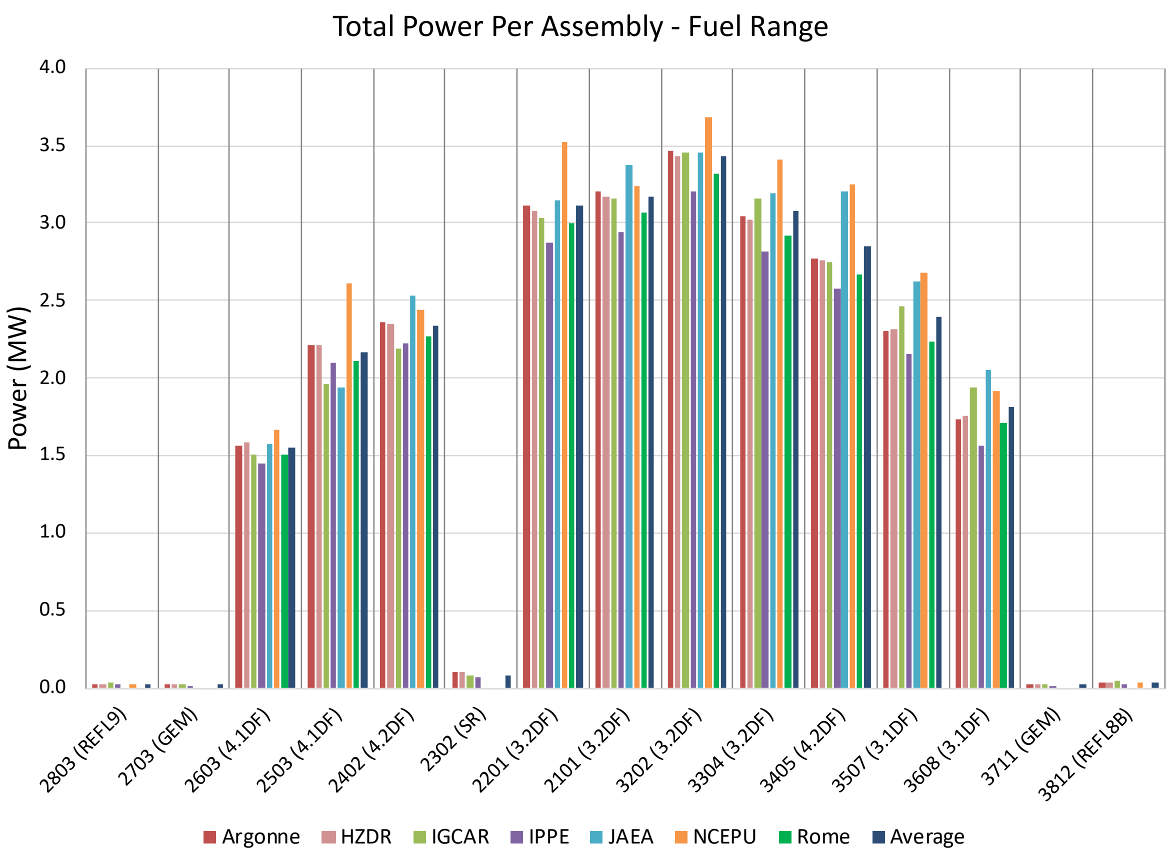

FIG. 3. Variation in Radial Power Profile In Several Assemblies

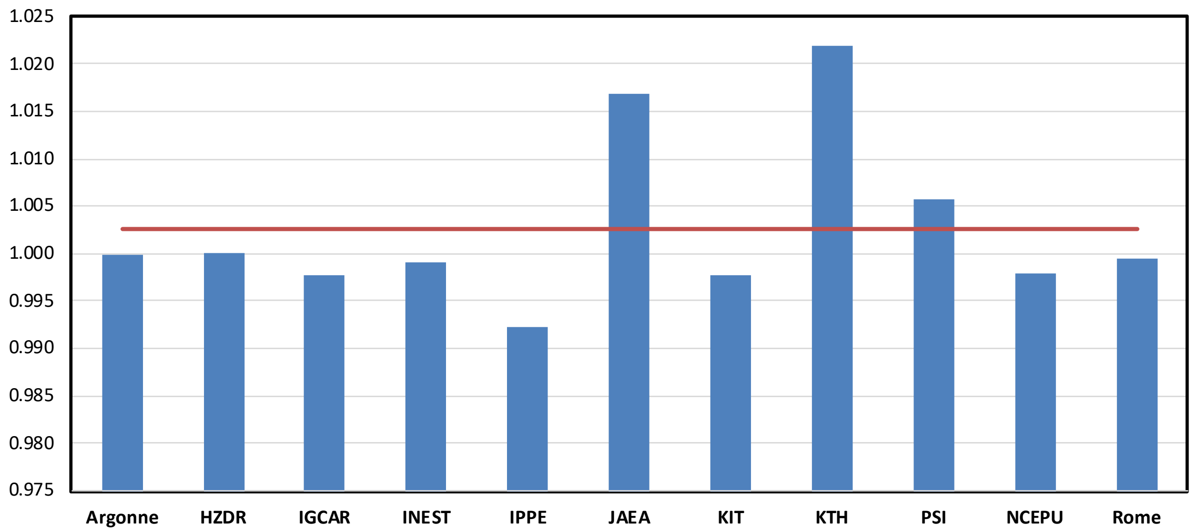

FIG. 4. Neutron Multiplication Factor and Average (Red Line)

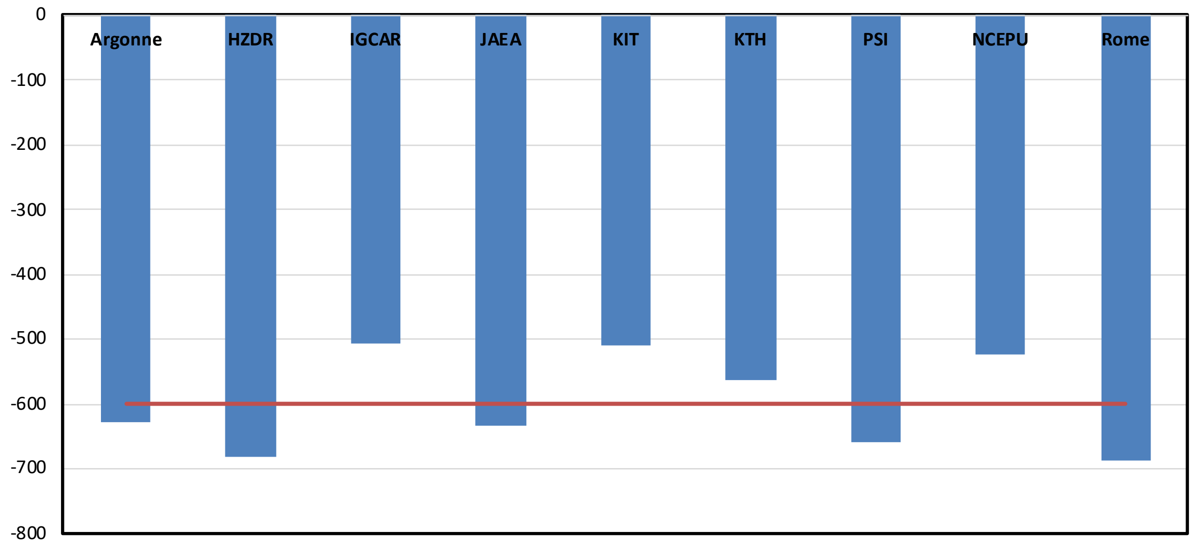

FIG. 5. Fuel Doppler Constant (pcm) and Average (Red Line)

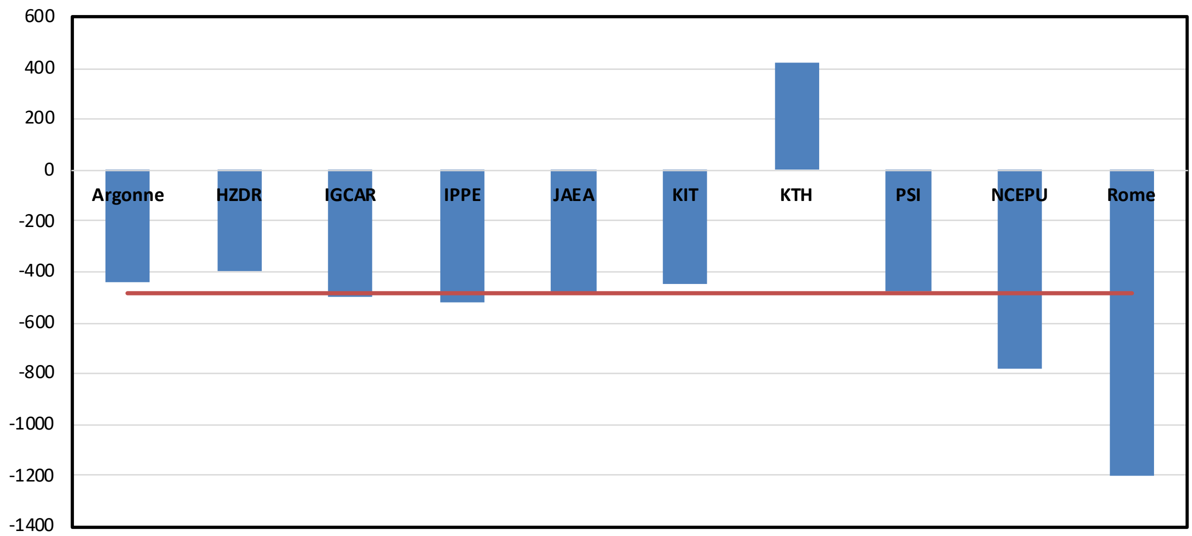

FIG. 6. Blind Results – Gas Expansion Modules (pcm) Worth and Average (Red Line)

4. CONCLUSIONS

The IAEA initiated a coordinated research project (CRP) in 2018 for the analysis of the Fast Flux Test Facility (FFTF) Loss of Flow Without Scram (LOFWOS) Test #13 that includes a neutronic benchmark of the core parameters used in safety models. The results from the blind phase of this neutronic benchmark display relatively good agreement for most of the parameters, but also highlight some outliers that will be investigated through the open phase of the benchmark that started in October 2020. These results helped illuminate discrepancies in the reactivity estimates from the transient calculations presented in [5] for those participants who computed neutronic reactivity coefficients themselves based on the detailed reactor description provided in the benchmark specifications. In particular, future effort will focus on identifying the reasons for the remaining discrepancies observed to further improve participant’s confidence in their neutronic modeling approaches. To conclude, this CRP provides valuable benchmark exercise for participating organizations to verify their neutronics methods based on the Fast Flux Test Facility.

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TABLE 2. COMPILATION OF ALL NEUTRONICS BENCHMARK PARAMETER RESULTS

	ANL	HZDR	IGCAR	INEST	IPPE	JAEA	KIT	KTH	NCEPU	PSI	Rome	*σ/avg* †
Neutron Multiplication Factor	0.99996	1.00003	0.99772	0.99900	0.99230	1.01689	0.99765	1.02200	0.99787	1.00574	0.99956	*0.10%*
Delayed Neutron Fraction (pcm)	313.1	312.9	334.0	650.0	324.0	315.7	364.0	341.0	375.0	320.9	300.0	*7%*
Prompt Neutron Lifetime	5.260E-07	5.429E-07	4.780E-07		5.65E-07		5.88E-07	6.30E-07		5.52E-07	4.820E-07	*9%*
Axial Expansion Coefficient (pcm/°C)	-0.322	-0.335	-0.227			-0.319		-0.300	-0.096	-0.221	-0.477	*36%*
Radial Expansion Coefficient (pcm/°C)	-1.000	-1.411	-1.220			-0.997		-0.930	-0.945	-1.522	-5.866	*19%*
Fuel Doppler Constant (pcm)	-629.0	-682.0	-507.5			-634.3	-509.0	-564.0	-524.3	-657.7	-687.7	*12%*
Fuel Density Coefficient (pcm/°C)	-1.362	-1.389	-1.450			-1.362		-1.360	-0.092	-1.363	-1.402	*1%*
Structure Density Coefficient (pcm/°C)	-0.121	0.219	0.200			0.093		0.100	-0.007	0.039	-0.098	*221%*
Sodium Density Coefficient (pcm/°C)	-0.346	-0.759	-0.912			-0.413	0.094	-0.940	-0.041	-0.274	-1.914	*81%*
Control and Safety Rods (pcm)	-11849	-10864			-9396	-10800		-11540	-8343	-11823	-12773	*12%*
Gas Expansion Modules (pcm)	-442	-394	-498		-516	-489	-448	420	-782	-475	-1201	*8%*

† - The estimate of the standard deviation divided by the average excludes the highlighted outliers. The outliers were confirmed as outside the average +/- 2-σ after exclusion.