Time-Dependent External Distributed Neutron Source Capability in PARCS and Comparison with TORT-TD

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Abstract – *This paper describes the extension of the hexagonal TPEN solver of the PARCS code to enable 3-D transient simulations involving time-dependent distributed neutron sources. The implementation aims at the safety assessment of subcritical fast spectrum systems driven by an external neutron source (ADS, Accelerator-Driven Systems). The PARCS code extensions are based on the methodology and experience gained from TORT-TD and verified by means of an OECD/NEA benchmark by comparisons with corresponding TORT-TD calculations. In addition, preliminary results of a coupled neutron-kinetics/thermal-hydraulics simulation of a neutron source transient in a lead-bismuth cooled fast spectrum system using PARCS/ATHLET is shown.*

I. INTRODUCTION

The safety assessment of subcritical systems driven by an external neutron source (ADS, Accelerator-Driven Systems) requires the application of 3-D neutron kinetics codes which can account for time-dependent distributed external neutron sources. An example of an ADS is represented by the MYRRHA facility which is designed as a fast spectrum system to demonstrate the physics and technology of an ADS for transmuting long-lived radioactive waste [1]. The hexagonal TPEN diffusion solver of the PARCS code [2] has recently been extended by GRS to also enable 3-D transient simulations involving time-dependent distributed neutron sources. Together with models to simulate radial and axial core expansion/contraction effects [3], this PARCS extension represents an important means for a future comprehensive safety assessment tool for fast critical and subcritical reactor systems. This paper describes the verification of the newly implemented external neutron source simulation capability into PARCS by means of an OECD/NEA benchmark and comparison with corresponding TORT-TD calculations.

II. IMPLEMENTATION OF THE TIME-DEPENDENT DISTRIBUTED NEUTRON SOURCES CAPABILITY IN PARCS

The implementation of the external neutron source capability into PARCS is based on the methodology and experience gained from the time-dependent transport code TORT-TD [3], which has been successfully extended by this feature [5] and verified on the YALINA-Thermal experiment [6] a few years ago. For the implementation in PARCS, the TPEN diffusion solver for hexagonal geometry has been selected, because the majority of fast neutron reactors use hexagonal assembly core layouts.

In standard notation [7], the time-dependent multi-group diffusion equation reads:

$$\left[\frac{1}{v_g}\frac{\partial}{\partial t} - \vec{\nabla} D_g(\vec{r})\cdot\vec{\nabla} + \sigma_g^{tot}(\vec{r})\right]\phi_g(\vec{r},t) = q_g(\vec{r},t) + \sum_{g'}\sigma_{gg'}^{scatt}(\vec{r})\phi_{g'}(\vec{r},t) + \\ + \chi_g(1-\beta)\sum_{g'}\upsilon\sigma_{g'}^{fiss}(\vec{r})\phi_{g'}(\vec{r},t) + \sum_l \chi_{gl}^d \lambda_l c_l(\vec{r},t)$$

Therein, $q_g(\vec{r},t)$ represents the time-dependent external neutron source term which is a function of energy group g and spatial location $\vec{r}$. The time dependence of the source is given by sampling the source at arbitrary time points. These time points need not to agree with the problem time steps of the PARCS transient simulation; source values at required problem time points are obtained by a linear interpolation of source values between given sampling points.

III. BENCHMARK PROBLEM DESCRIPTION

The verification of the external neutron source simulation capability of PARCS has been carried out by solving the OECD/NEA benchmark “Comparison Calculations for an Accelerator Driven Minor Actinide Burner” [8]. This benchmark defines a simplified cylindrical model of an ADS reactor core used to burn plutonium and minor actinides whose general layout is reproduced in Fig. 1. The core consists of the following four homogenized material zones: target, fuel, reflector and a void zone where the proton beam enters the core. The operating temperatures are 980 K for the fuel and 650 K for the other materials. The core power is 377 MW, and the fuel composition is considered at its start-up state. Further details can be found in the benchmark report [8].

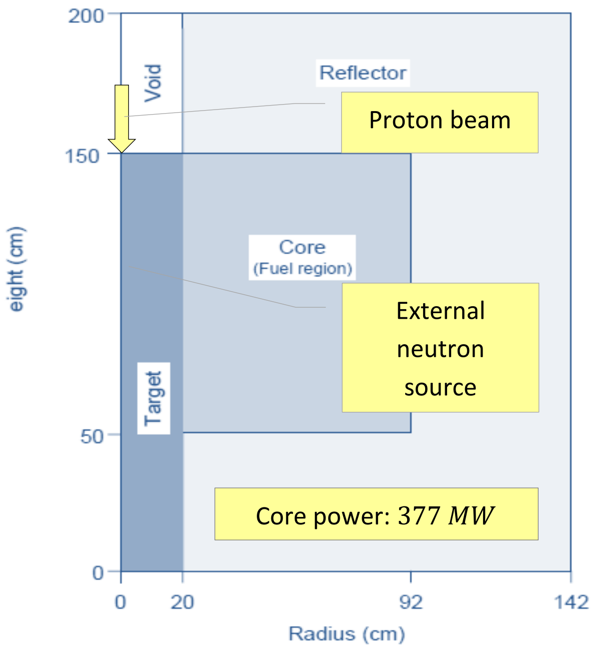

Fig. 1: *r-z* model of the simplified accelerator-driven minor actinide burner system (modified from [8]).

Tab. I: 8-group energy structure used for PARCS and TORT-TD.

Group index	Lower energy boundary (MeV)	Group index	Lower energy boundary (MeV)
1	2.23000E+00	5	4.09000E–02
2	8.21000E–01	6	1.50000E–02
3	3.02000E–01	7	7.49000E–04
4	1.11000E–01	8	1.00000E–10

Tab. II: Neutron energy groups velocities.

Group index	Neutron velocity (cm/s)	Group index	Neutron velocity (cm/s)
1	2.49296E+09	5	3.60202E+08
2	1.53190E+09	6	2.16628E+08
3	9.52181E+08	7	1.03564E+08
4	5.89654E+08	8	3.18034E+07

Tab. III: Delayed neutron parameters.

Prcursor family index	$\boldsymbol{\beta_i}$ **(pcm)**	$\boldsymbol{\lambda_i\ (s^{-1})}$
1	185.74	4.10544E-01
2	3.76	1.04564E-02
3	47.55	2.82766E-02
4	34.49	1.01611E-01
5	66.92	2.99192E-01
6	28.05	9.54739E-01
7	4.97	3.12438E+00

IV. SIMULATION MODELS IN PARCS, TORT-TD AND SERPENT

1. Generation of Few-Group Macroscopic Cross-Sections

For the preparation of nuclear few-group data to be used in PARCS and TORT-TD, the 8 energy group structure shown in Tab. I has been applied. Homogenized cross sections for the different materials of the ADS benchmark core (fuel, target and reflector) were produced by the Monte-Carlo code Serpent [9] using 2-D infinite medium models and employing ENDF/B-VII based continuous energy nuclear data. The obtained neutron group velocities are listed in Tab. II, the delayed neutron parameters are found in Tab. III. Serpent was also used to generate a 3-D reference solution for the full cylindrical problem geometry. In both full-core and cross section generation models, 2000 neutron histories, 20 inactive and 480 active cycles were used. Most of the generated homogenized nuclear data for the 8-energy group structure had statistical uncertainties below 1 %.

2. Deterministic Models for PARCS and TORT-TD

In TORT-TD, the benchmark problem can be directly modeled in cylindrical coordinates. The corresponding analyses were performed using both the S_N neutron transport method and the diffusion approximation to compare with the PARCS diffusion solutions. As PARCS does not have a solver that operates with cylindrical coordinates, and as the time-dependent external neutron source capability has been implemented for the (hexagonal) TPEN solver, it was necessary to transform the cylindrical core into an equivalent hexagonal geometry under the constraint that the material zone volumes of the cylindrical core are preserved in the hexagonal model. This leads to the core configuration shown in Fig. 2. In this model, the target is represented by 7 assemblies. Further parameters of this equivalent model are listed in Tab. IV.

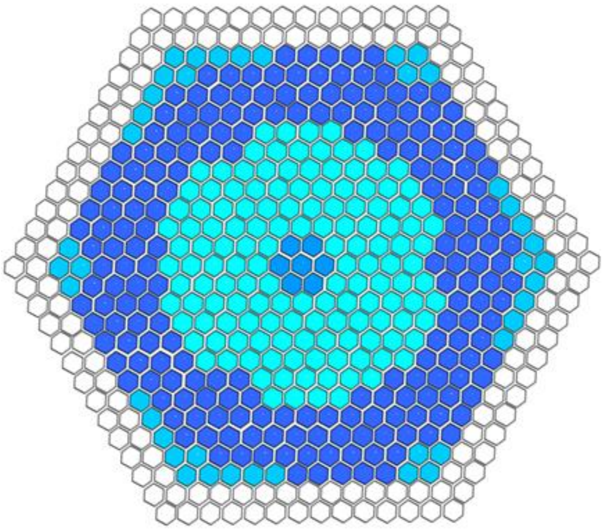

Fig. 2: Hexagonal radial core configuration equivalent to the cylindrical core.

Tab. IV: Parameters of the equivalent hexagonal core model.

Parameter	Value
Number of target assemblies	7
Number of fuel assemblies	141
Number of reflector assemblies	207
Assembly flat-to-flat pitch	14.3976 cm

3. Spallation neutron source description

A spallation neutron source is provided with the OECD/NEA benchmark specification in terms of a multi-group source spectrum and an axial distribution profile within the target zone adjacent to the active core region (see Tab. V). A source collapsing program is also provided along with the benchmark specification, which has been used to reduce the spallation group structure from the original 122 to the 8 energy groups given in Tab. I.

Tab. V: Relative axial distribution of the spallation neutron source.

Axial position (cm)	Source intensity (%)	Axial position (cm)	Source intensity (%)
50-60	33.369	100-110	3.1930
60-70	26.611	110-120	1.1310
70-80	17.754	120-130	0.5710
80-90	10.825	130-140	0.3000
90-100	6.0850	140-150	0.1610

For source transient calculations, a rectangular neutron source pulse has been defined as shown in Fig. 3. The source, being already fully switched on during steady state, is switched off for a duration of 9 s, switched on again for the next 60 s and afterwards reduced to half of the full intensity.

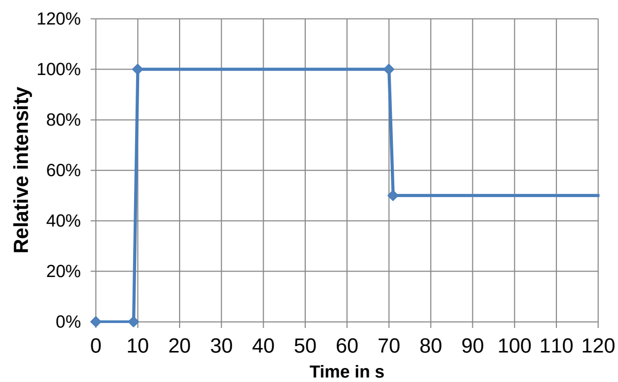

Fig. 3: Time-dependence of the external neutron source strength.

V. SIMULATION MODEL VERIFICATION

The verification of the deterministic simulation models for PARCS and TORT-TD is done in two steps. First, the few-group cross section data is verified by means of the cylindrical TORT-TD model and comparison with the Serpent Monte Carlo full-core results. Second, the equivalence of the hexagonal PARCS model is shown by comparison with TORT-TD full-core results.

1. Comparing TORT-TD with Serpent

For the verification of the deterministic simulation models, steady-state results obtained with the cylindrical TORT-TD model and the nuclear data in 8 energy groups have been compared with the Serpent full-core model using continuous energy data, both without external neutron source. To this aim, TORT-TD transport calculations using S_4 and S_{24} angular quadratures and, in addition, a TORT-TD diffusion calculation have been carried out. The reason for obtaining a diffusion solution with TORT-TD is to compare later with PARCS, which is a diffusion code. As can be seen from Tab. VI, the TORT-TD steady-state eigenvalues are very close to the Serpent full core reference solution; as expected, the diffusion approximation yields a larger deviation.

Tab. VI: TORT-TD steady-state eigenvalues compared to the Serpent full core reference solution.

Code / model	k_{eff}
Serpent	0.95998
TORT-TD S_8	+56 pcm
TORT-TD S_{24}	+20 pcm
TORT-TD Diffusion	+126 pcm

The radial and axial flux and power profiles of the TORT-TD simulation with diffusion approximation compared with the Serpent reference solution are given in Fig. 4 through Fig. 7. The radial and axial power profiles predicted by TORT-TD diffusion approximation are in acceptable agreement with the Serpent results with highest deviations of about 3.6% mostly at materials interfaces. Regarding the neutron flux profiles, the deviations from Serpent are up to 7% to 9% for the axial and radial profile in the fissile region, respectively. Larger errors are found in non-fissile regions, e.g. the reflector.

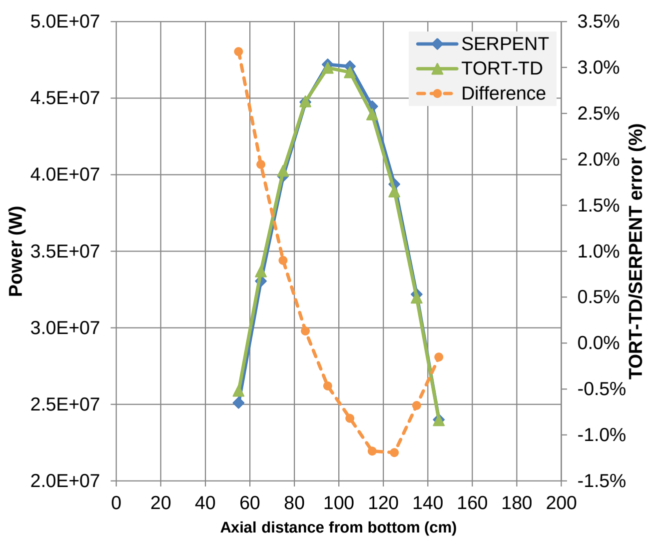

Fig. 4: Axial power profile of the TORT-TD diffusion solution compared with the Serpent reference simulation.

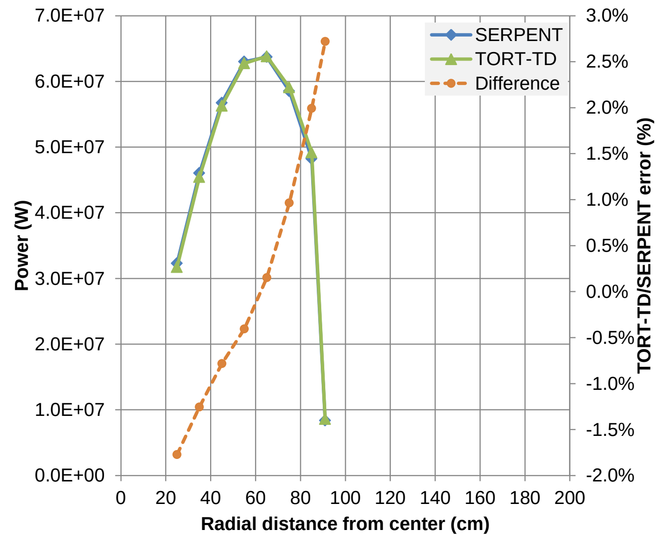

Fig. 5: Radial power profile of the TORT-TD diffusion solution compared with the Serpent reference simulation.

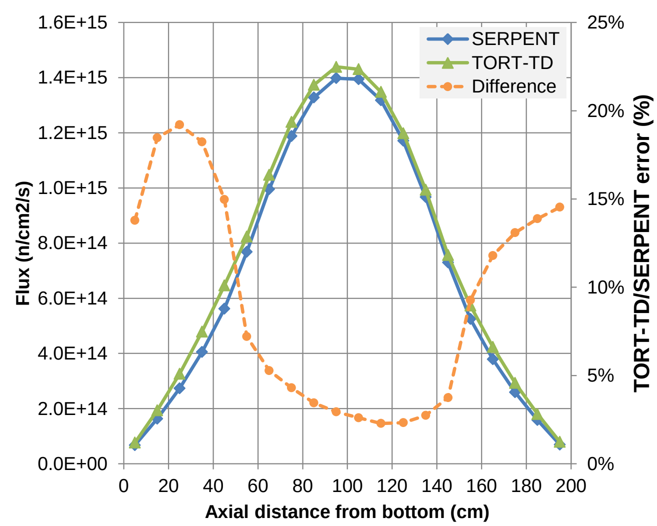

Fig. 6: Axial neutron flux profile of the TORT-TD diffusion solution compared with the Serpent reference simulation.

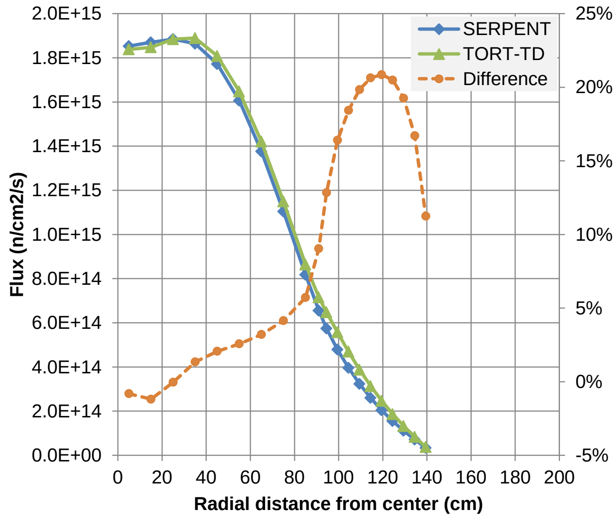

Fig. 7: Radial neutron flux profile of the TORT-TD diffusion solution compared with the Serpent reference simulation.

2. Comparing the hexagonal PARCS with the cylindrical TORT-TD model

In the second step, the same set of 8 energy group data has been used in the PARCS TPEN solver in combination with the equivalent hexagonal model. The hexagonal PARCS model yields a multiplication factor of 0.95580 which is 544 pcm below the cylindrical TORT-TD diffusion approximation eigenvalue. The radial and axial neutron flux distributions obtained with PARCS agree within less than 1.7 % with the cylindrical TORT-TD results (Fig. 8 and Fig. 9). This also applies to the radial and axial power distributions, where maximum differences do not exceed 1.5 % as can be seen from Fig. 10 and Fig. 11.

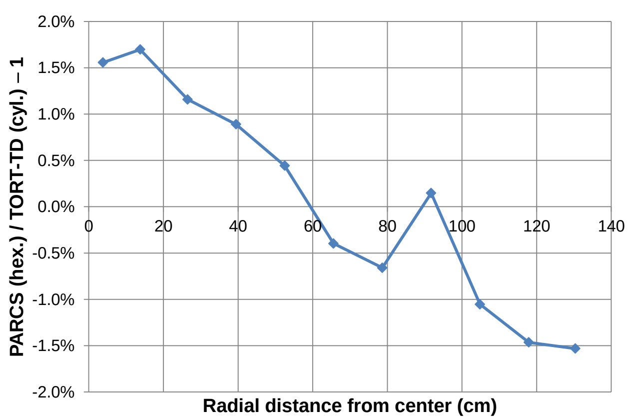

Fig. 8: Relative deviation of the hexagonal PARCS model radial flux distribution w.r.t. the cylindrical TORT-TD model.

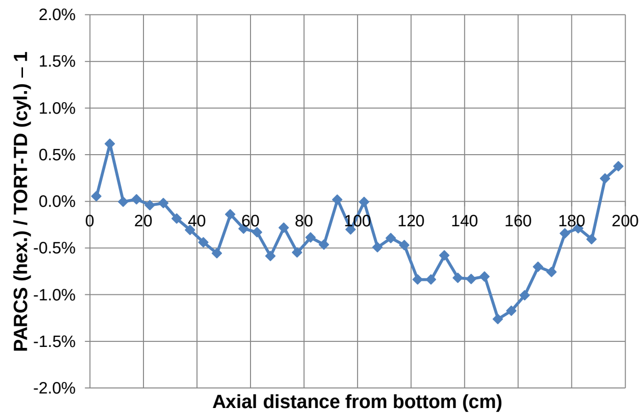

Fig. 9: Relative deviation of the hexagonal PARCS model axial flux distribution w.r.t. the cylindrical TORT-TD model.

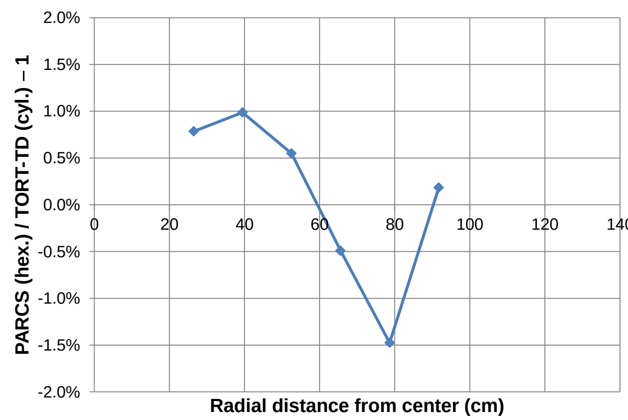

Fig. 10: Relative deviation of the hexagonal PARCS model radial power distribution w.r.t. the cylindrical TORT-TD model.

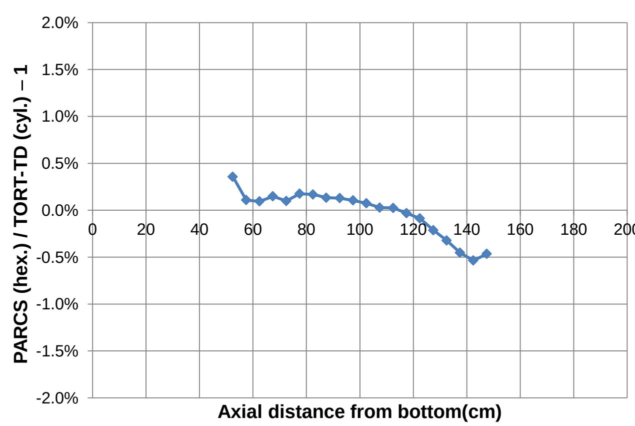

Fig. 11: Relative deviation of the hexagonal PARCS model axial power distribution w.r.t. the cylindrical TORT-TD model.

VI. SOURCE TRANSIENT SIMULATION RESULTS

Using the nuclear cross sections determined by Serpent and the neutron source pulse definition along with the axial source intensity profile, a source transient simulation with PARCS has been done with a constant time step size and compared with TORT-TD results.

The total power evolution during the source transient simulation is shown in Fig. 12. It can be seen that the power follows closely the rectangular source shape; during periods with constant source, the power evolution is due to the delayed neutron characteristics. In a second example with a much shorter neutron pulse, the rectangular source pulse duration was reduced to 0.4 s, with a 0.9 s source-free period at the beginning of the transient; in this case, no source is present at steady state. The corresponding PARCS and TORT-TD results are depicted in Fig. 13. Despite of the different core models in PARCS and TORT-TD (hexagonal and cylindrical, respectively), this demonstrates that the time-dependent external distributed neutron source capability is correctly implemented in PARCS.

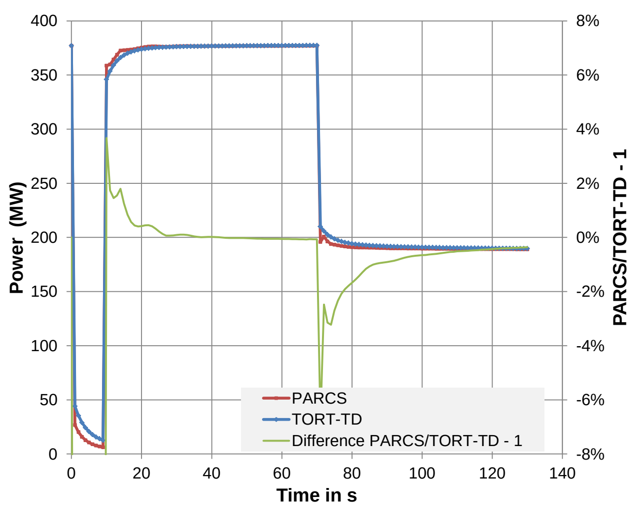

Fig. 12: Total power evolutions during the source transient obtained with PARCS (green line) and TORT-TD (blue line). The green line denotes the relative difference between both codes.

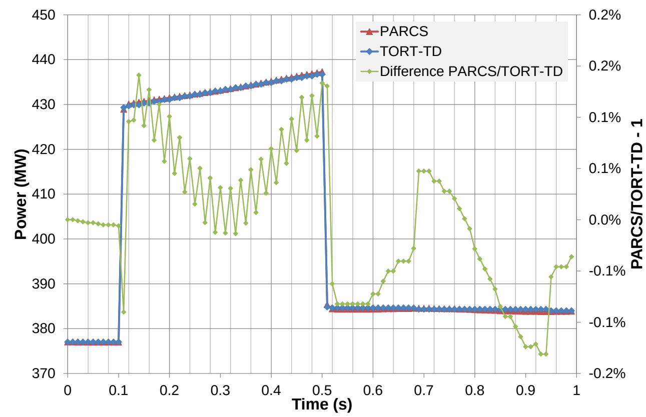

Fig. 13: Total power evolutions during the source transient obtained with PARCS (green line) and TORT-TD (blue line). The purple line denotes the relative difference between both codes.

The last example depicted in Fig. 15 shows preliminary results of an application of the coupled code system PARCS/ATHLET [10] for the simulation of a source transient (rectangular pulse of 5 s duration starting at 120 s) in a lead-bismuth cooled core based on MYRRHA specifications [11]. For this core, which is shown in Fig. 14, a parallel channel thermal hydraulic model with the GRS system code ATHLET [12] has been developed. For thermal hydraulic feedback, a parameterized macroscopic cross section library in 8 energy groups has been prepared with Serpent which is parameterized with respect to fuel temperature and coolant density. It can be seen that negative thermal-hydraulic feedback effects limit the power excursion initiated by the external neutron source pulse.

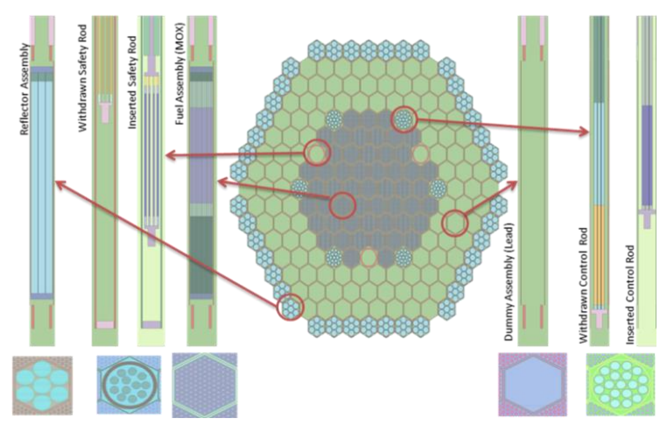

Fig. 14: Serpent model of the MYRRHA core (center) showing axial and radial subassemblies cross sections.

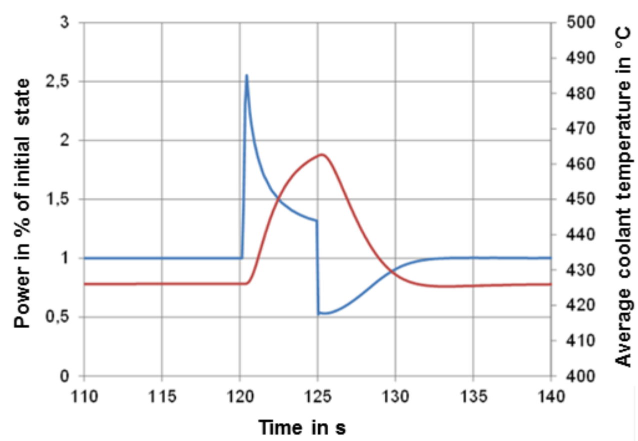

Fig. 15: Coupled PARCS/ATHLET simulation of a source transient in a MYRRHA core (preliminary). The blue line shows the power evolution, the red line the evolution of the average coolant temperature.

VII. SUMMARY AND CONCLUSION

This paper presents extensions to the hexagonal TPEN solver of the PARCS code to enable 3-D transient simulations involving time-dependent distributed neutron sources. Based on an OECD/NEA benchmark for a simplified ADS core configuration, comparison calculations between PARCS and TORT-TD are shown in order to verify the newly implemented external neutron source simulation capability of PARCS. To this aim, two transient cases with different spallation neutron source pulse durations, 60 s and 0.4 s, have been simulated. Macroscopic cross sections have been generated with the Serpent Monte Carlo code in 8 energy groups, which has also been used as reference for the steady state whole core flux distributions. The transient results are physically reasonable and compare well with the TORT-TD solution, so providing confidence in future application of PARCS to source driven fast neutron spectrum systems, also coupled with thermal hydraulic codes.

ACKNOWLEDGEMENT

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