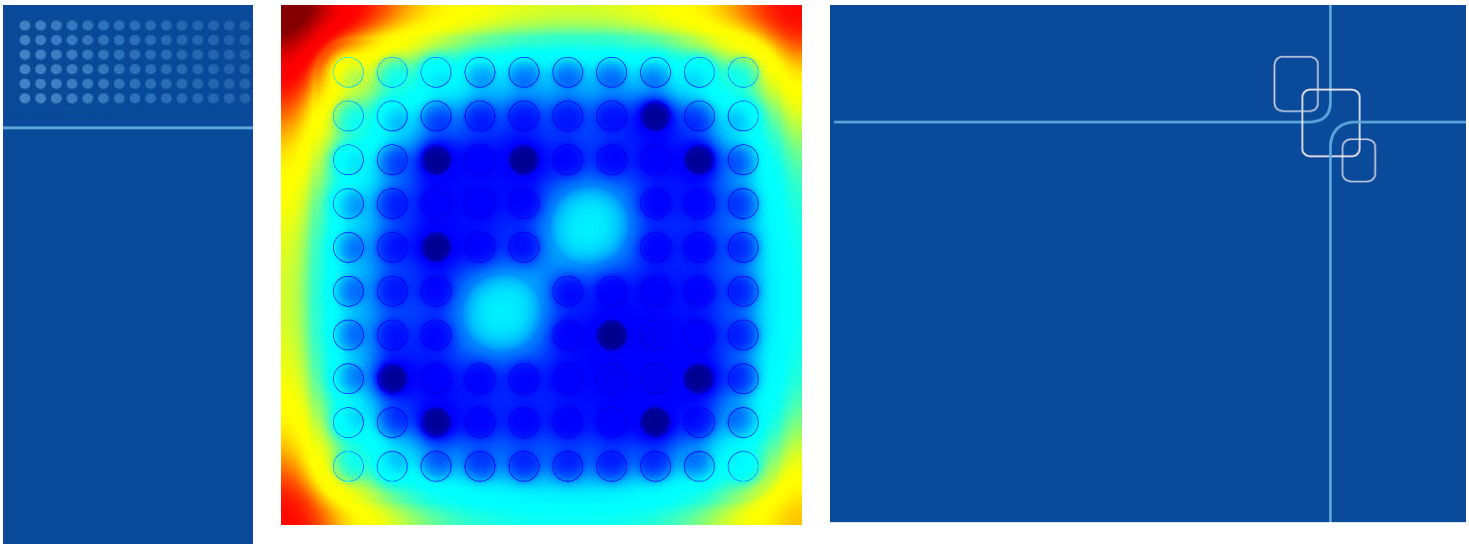
RESEARCH REPORT

VTT-R-05444-12



Serpent Progress Report 2011

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Summary

This report summarizes the work carried out for the development of the Serpent Monte Carlo code during year 2011.

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Contents

1 Introduction

The Serpent Monte Carlo code has been developed at VTT Technical Research Centre of Finland since 2004, for the purpose of reactor physics applications. The code is specialized in assembly-level calculations, including depletion studies and the production of homogenized multi-group constants for full-core reactor simulator codes. The field of applications has been broadened over the years, and the code is currently also used for modeling research reactors and other complicated three-dimensional systems. Future work is mainly focused on a new code version, Serpent 2, with the purpose of extending the scope of applications into multi-physics and radiation shielding, and taking full advantage of high-performance parallel computing.

This report summarizes the work carried out for the development of Serpent 1 and Serpent 2 in the year 2011. A more complete description of the capabilities and applications is found at Serpent website – http://montecarlo.vtt.fi.

2 Background

The development of Serpent was started at VTT in 2004, under the working title Probabilistic Scattering Game, or PSG [1]. The name Serpent has been used since October 2008, when a per-release version was distributed to some research institutes for testing purposes. The code has been in public distribution at the OECD/NEA Data Bank since May 2009 and RSICC in the U.S since March 2010. The code is licensed for non-commercial research and educational use.

The Serpent website was established in October 2008, and it still serves as the best and the most up-to-date description of code status and capabilities. The website also contains the most recent version of the User's Manual, descriptions of new updates, a list of Serpent-related publications, example input files and a list of recent and up-coming events. An interactive web-based discussion forum was set up for the user community in March 2010 at http://ttuki.vtt.fi/Serpent.

Serpent development has received full support from VTT since the beginning, and the work has been mainly funded from the reactor physics projects in the Finnish National Research Programs on Nuclear Power Plant Safety (SAFIR and SAFIR-2010). The work carried out in 2011 was mainly funded from the KÄÄRME project in the new SAFIR-2014 research program, and the EU High Performance Monte Carlo Reactor Core Analysis (HPMC) project. The developer team at VTT consists of three scientists, with significant contribution from universities and research institutes in Finland and abroad.

3 Code development in 2011

Serpent 1 source code was updated from version 1.1.14 at the beginning of the year to 1.1.16 in September 2011. The main new features and major bug fixes are

described below and more detailed description can be found at the Serpent website. Development of Serpent 2 is discussed separately in Section 3.2.

3.1 New features and major bug fixes

During 2011 the main focus in code development was shifted to Serpent 2. The source code for Serpent 1 is still maintained and updated, but the work is limited to bug fixes and minor new features. The first international Serpent user group meeting, held in September 2011, brought up several requests from the user community for improving the existing capabilities. Most of these requests involved the calculation of new multi-group constants, improvements in boundary conditions and the capability to tally surface currents [2]. Some of the work was still in progress in 2012.

Most of the bug fixes were relatively minor. The physics routines were revised while developing Serpent 2, which lead to the discovery of several systematic errors in some of the ENDF scattering laws at high energy. The corrections were also implemented in Serpent 1, but the impact on results remained small since most of the errors affected rare scattering events at energies above the fission spectrum.

3.2 Development of Serpent 2

The development of Serpent 2 started in September 2010, and the basic routines performing transport and burnup calculations were completed by June 2011. The code is currently in a beta-testing phase, started in January 2012. The source code is available to licensed Serpent users and public distribution is scheduled for 2013-2014.

The main reason for starting a new code version was that the calculation methods in Serpent 1 were becoming increasingly complicated, making it extremely difficult to implement any major new capabilities. Serpent 1 also suffers from an appetite for computer memory, which becomes a major limitation in large-scale burnup calculation problems. It became clear that with this excessive memory usage, together with parallelization based on distributed-memory MPI, it would eventually become impossible to keep up with computer development, heading more and more in the direction of multi-core CPU's and massive parallelization.

One of the main goals for Serpent 2 is to extend the burnup capability from 2D assembly-level calculations all the way to full-core problems consisting of hundreds of thousands of depletion zones, without any limitations in parallelization. Most part of 2011 was spent developing new parallelization routines based on the combination of OpenMP and MPI, and different optimization modes for small and large burnup calculation problems [3].

Serpent 2 has also become the main platform for developing new features. Work began in 2011 for the development of gamma and coupled neutron-gamma transport capability. One of the main areas of interest at the moment is multi-physics, which involves coupling Serpent 2 to thermal hydraulics and fuel performance codes. Some preliminary work, involving an on-the-fly temperature treatment routine was started in 2011 [4, 5]. Other new features include advanced

time-integration methods for burnup calculation [6, 7] and non-analog Monte Carlo with simple variance reduction techniques.

4 User community

The number of user organizations grew from 37 in January 2011 to 57 by the end of the year. The number of registered users increased to about 140.

The first international Serpent user group meeting was hosted by the Helmholtz-Zentrum Dresden-Rossendorf in Dresden, Germany, in September 15-16, 2011. The meeting brought together 33 Serpent users from 15 Organizations around the world. The first day of the meeting focused on applications and methods used in Serpent 1. The second day began with an introduction to Serpent 2, followed by presentations related to new features and capabilities. The afternoon of the second day was reserved for discussion. The presentations are available for downloading at website: http://montecarlo.vtt.fi/mtg/2011_Dresden/index.htm.

The applications for the Serpent code have ranged from group constant generation and fuel cycle studies to research reactor calculations. Some example applications and other activities in 2011 are described below.

4.1 Paul Scherrer Institut (PSI)

Serpent has been used in two different groups of the Laboratory of the Reactor Safety (LRS) of the Paul Scherrer Institut. Below are described the various activities performed in the FAST (Fast-spectrum Advanced Systems for power production and resource management) group and STARS (Steady-state and Transient Analysis Research for the Swiss Reactors) group of the LRS.

4.1.1 FAST Activities

In framework of an on-going doctoral study at PSI, Serpent 1.1.16 has been used to verify the sodium-cooled fast reactor calculations carried out by the deterministic code ERANOS. In particular, the neutron spectrum and the fission macroscopic cross-section (in an infinite fuel lattice) are compared. It should be mentioned that ECCO is adopted in ERANOS for the lattice calculation, which generates 1968-group fluxes and 33-group cross-sections; whereas Serpent predefines the same energy grids as ECCO in order to generate the comparative group constants.

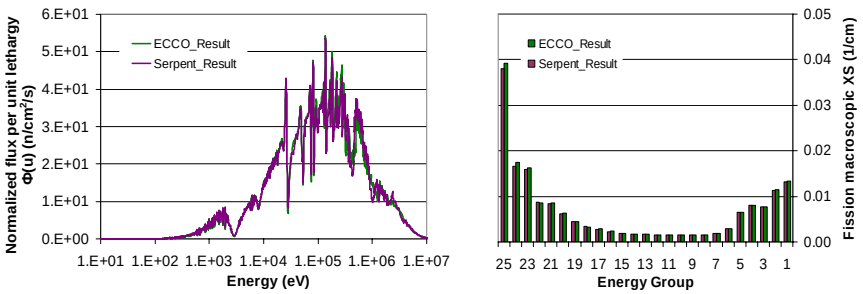

Figure 1. Normalized neutron spectra (left) and group-wise fission macroscopic cross-sections (right) generated by ERANOS/ECCO and Serpent, respectively.

It is found that the two neutron spectra generated by ECCO and Serpent are in a very good agreement (see Figure 1 left). Furthermore, the predictions of fission macroscopic cross-section at the energy range above 60 eV (i.e. above energy group No.25) are quite consistent between the two codes (see Figure 1 right). Large discrepancy, however, can be found at the lower energy range below 60 eV (i.e. below energy group No.25). The observed discrepancy can be explained by not enough counts being accumulated in Serpent at the low energy regions, so that the statistical error is significant there. Nevertheless, for a study focusing on the fast spectrum system, the results are considered satisfactory.

4.1.2 STARS Activities

One objective of STARS is to move towards the capability to prepare few group homogenized macroscopic cross sections for downstream 3-D core simulators like SIMULATE-3, S3K and/or PARCS with the Serpent code. The goal of the present work is to perform a preliminary assessment of a PARCS calculation scheme where the macroscopic cross sections library is generated by Serpent for a beginning-of-life isothermal mini-core with comparison with the conventional CASMO-5/PARCS approach.

The mechanics to use Serpent as a lattice code, i.e. the scripts SerpentXS[1] and the GenPMAXS code have been installed. In terms of cross sections generation, it is hard to make a statement on the benefits of using Monte Carlo-generated few group cross sections for the 2D layout, typical of LWR reactors, considered. Due to the statistical uncertainties in the Monte-Carlo approach and nuclear data inconsistencies, the agreement between CASMO-5 and Serpent is within 200pcm at the assembly level. At the 2D mini-core level, no bigger difference is observed when comparing PARCS run with CASMO-5 and Serpent cross sections. For 3D cases, non-trivial differences are observed in the axial power prediction as seen in Figure 2 due to discrepancies in the diffusion coefficients. The issue is related to Serpent, has been reported before and ways to fix this problem exists and have been implemented in Serpent.

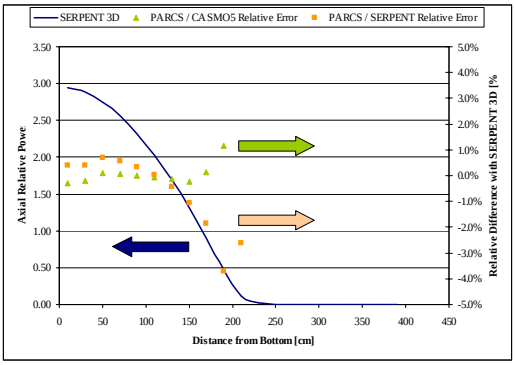

Figure 2. Comparison of the axial power variation between PARCS / CASMO-5, PARCS / Serpent and Serpent for a partially inserted control rod.

[1] SerpentXS is a python script written at MIT for the automated cross section generation using Serpent. The script is available for download at: http://canes.github.com/SerpentXS/

4.2 Lappeenranta University of Technology

Serpent was used mainly for calculating power distributions of pebble-bed type reactors. Serpent 1.x versions were used for a Master's thesis work where reactor physics were coupled with thermal hydraulics in a pebble-bed reactor. Reactor used in the calculations was a simple cylindrical pebble-bed with 100 000 pebbles and a central reflector. Locations of the single pebbles in the pebble-bed were calculated with in-house DEM (Discrete Element Method) -code and fuel particles were explicitly positioned in the pebbles.

Total amount of pebbles in different temperatures were limited to about 40 because of a memory limitations with MPI parallelization of Serpent 1.x. With a new threaded Serpent 2.x well over a hundred temperature values could be used in the same calculations. Thermal hydraulics were calculated with ANSYS Fluent using three-dimensional porous model with the porosity calculated from locations of the pebbles. Calculation data transfer between the two programs was done with Perl script which read pebble-wise power distribution calculated by Serpent and divided power in to calculation cells of Fluent. After a thermal hydraulic iteration the script built new input for Serpent with new temperatures. One such a Serpent plot with temperature distribution visible is presented in Figure 3.

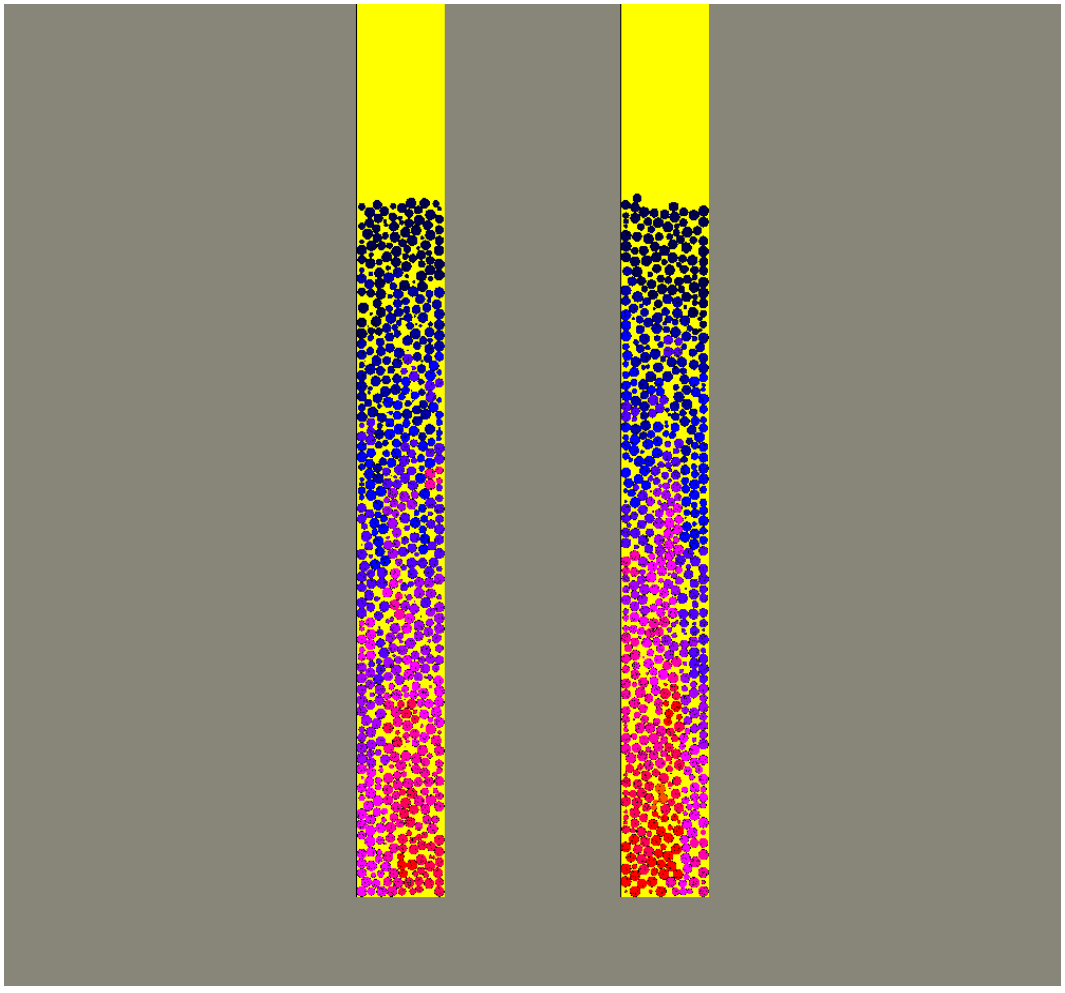

Figure 3. Serpent pebble-bed reactor model with randomly positioned pebbles and temperatures calculated with CFD (Computational Fluid Dynamics) using porous model.

4.3 Idaho National Laboratory (INL)

Researchers at the Idaho National Laboratory (INL) have begun testing the capabilities of Serpent while developing benchmarks for the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* and the *International Handbook of Evaluated Reactor Physics Benchmark Experiments*. The handbooks are used to preserve experimental data and measurements for use in the validation of computational and analytical methods and the refinement of integral nuclear data.

Calculations with Serpent 1.1.17 compare well with MCNP5-1.60 calculations and help provide confidence in MCNP results when there are discrepancies between MCNP and KENO-VI (SCALE 6.1) calculations when both are using continuous energy cross section data. Sample calculations using Serpent will be included for some benchmark reports in future editions of the handbooks. Current examples include the Neutron Radiography (NRAD) Reactor at INL (Figure 4), a small compact critical assembly (SCCA) from Oak Ridge National Laboratory (ORNL, Figure 5), and the GROTESQUE experiment from ORNL (Figure 6).

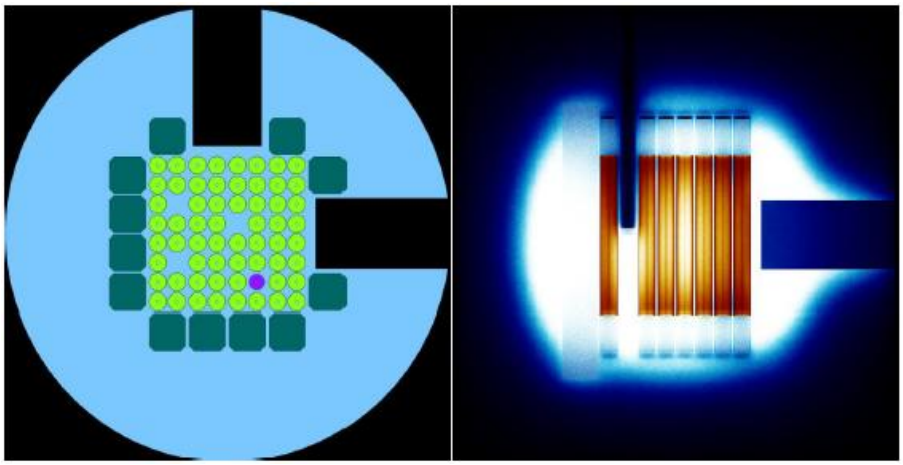

Figure 4. Serpent images of the NRAD Reactor at the INL, which is a graphite-reflected UErZrH-fueled TRIGA reactor.

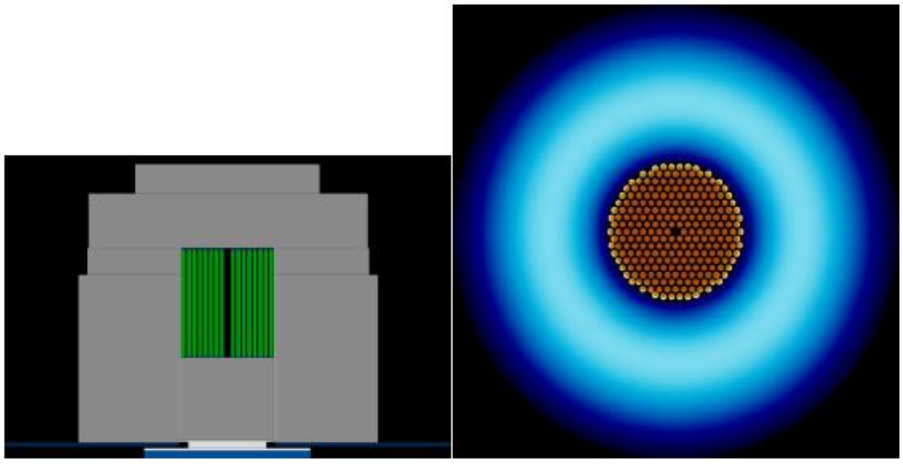

Figure 5. SCCA-002 experiment at ORNL in the 1963. It is an array of HEU-O2 reflected by graphite.

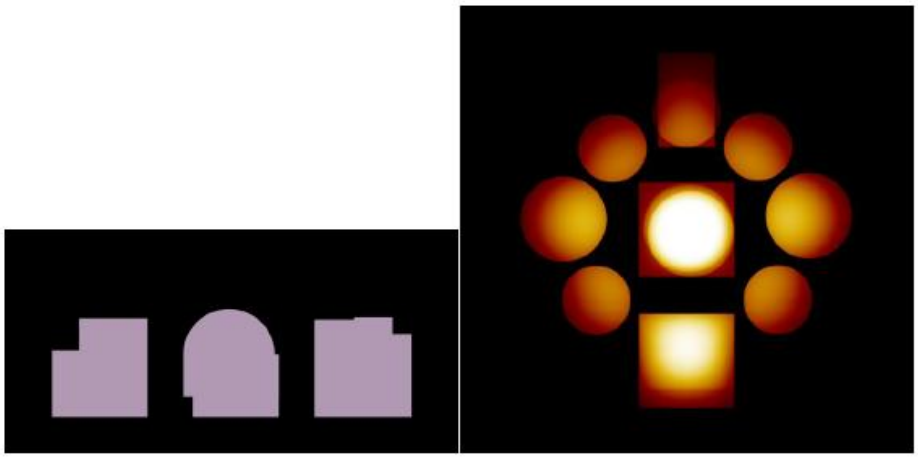

Figure 6. GROTESQUE experiment performed at ORNL in 1964. It is a critical assembly constructed from nine HEU metal units of varying dimensions.

4.4 Texas A&M University

Texas A&M University's 1MW TRIGA Mark I (Figure 7) allows for experimental validation of the Serpent code. The beginning of life (BOL) characteristics are very well known for this reactor since a fresh fuel loading was installed in 2006 followed by a detailed startup plan. BOL approach to criticality was completed with the Serpent model and compared to experimental rod heights (Table 1, shown by percent withdrawn). Reactor kinetics parameters are also compared to experimentally determined values. The burnup capability was used as well to generate isotopics for MCNP dose rate calculations from fuel bundles.

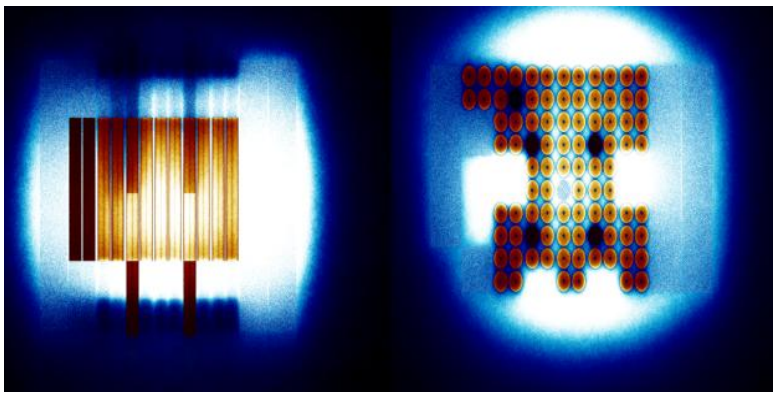

Figure 7. Visualization of thermalization and fission rate densities in the generic Serpent mesh for the TRIGA reactor in the XZ (left) and XY (right) planes.

Table 1. Critical control rod heights and reactor kinetics parameters for the TRIGA reactor.

Parameter	Experimental	Calculated
Transient Rod	100	100
Shim Safeties	49	49.8
Regulating Rod	45.6	45.6
k_{eff}	1.0	1.00014 +/- 0.00006
l_p	0.026178 ms	0.06967 ms

Serpent was also used to determine end of life (EOL) fuel parameters for use in dose calculations in MCNP. Flux measurements done at the NSC will be compared in future iterations of the model development

The Serpent model is also coupled with CFD simulations with Cd-Adapco's STAR-CCM+ and OpenFOAM codes. These couplings are done through a Java interface with cell splitting in Serpent. Power profiles are fitted with a fourth-order polynomial prior to implementation within the CFD code.

4.4.1 Education

Serpent was used in the classroom to study prismatic-block helium-cooled graphite-moderated reactors (Figure 8). In particular, Serpent was used to evaluate the core lifetimes for various TRISO particle radii, packing fractions, and fuel types. Serpent was chosen for these burnup calculations due to its superior calculation speed and ease of use relative to other Monte Carlo depletion codes. While the full core was modeled, 1/6th core symmetry was utilized when defining the burnable regions.

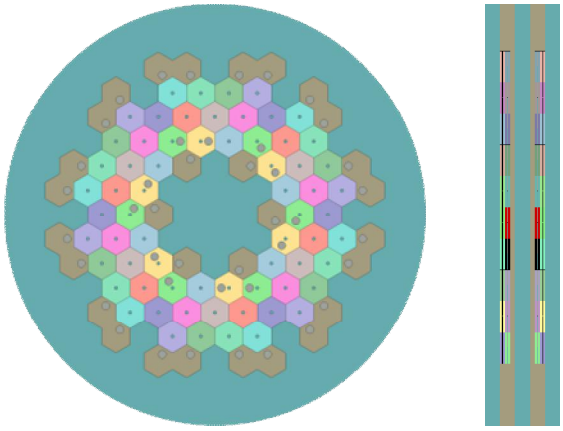

Figure 8. Two views of the core geometry that depicts how burnable materials were assigned. An x-y slice is shown on the left, while an x-z slice is shown on the right. Each color represents a different homogenized burnable region for visualization purposes.

The double heterogeneity of the fuel compacts was modeled explicitly in Serpent. In reality, TRISO particles would be randomly packed inside each pin; however, they were modeled with a square lattice.

4.4.2 Advanced Energy Technologies Research Group

Reactor concepts were developed in Serpent since fuel utilization and characterization of core life behavior is fairly painless with Serpent. In the Advanced Energy Technologies (AET) group, novel and complex reactor concepts are evaluated. These often require Monte Carlo techniques since the material and geometry constraints are difficult to resolve elsewhere.

High Temperature Integrated Multi-Modular Reactor [8, 9]

Serpent was used to develop a novel small modular reactor design (High Temperature Integrated Multi-Modular Thermal Reactor, Figure 9). Sensitivities of the core multiplication factor and lifetime on fuel enrichment, geometry, and control/safety elements were studied. Flux and power distributions were obtained and used in simple thermal hydraulics hot channel analyses. The core was made into a fast reactor design as well as a thermal reactor design. The work culminated in a M.S. thesis.

A multi-modular core was created. Each module consisted of a subcritical fuel assembly and balance of plant machinery. A critical configuration was found when several modules were put within proximity of one another. Studies on the usefulness of this modular configuration were performed using Serpents relatively fast burnup calculation times.

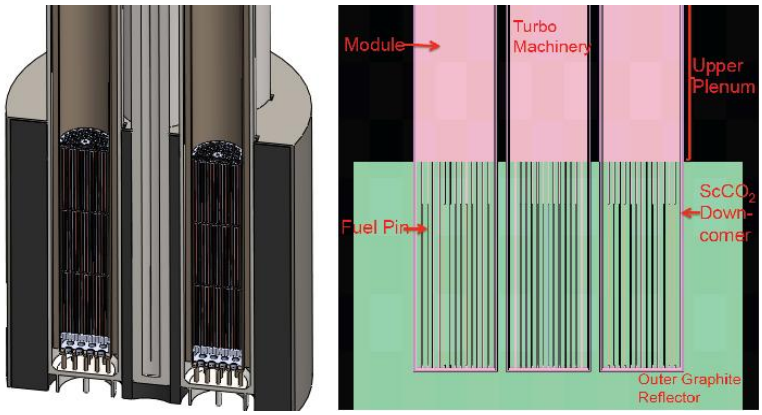

Figure 9. SolidWorks (dimensions are arbitrary for visualization) and Serpent geometry representation of conceptual reactor design.

The reactor used an enhanced thermal conductivity BeO-UO2 fuel. The effects of BeO additive to UO2 on core lifetimes was performed using Serpent's burnup functionality (Figure 10). This study concluded that a 5% BeO additive would be a good basis to do calculations with. The final fuel form would be decided upon many other parameters. The radial core flux distribution was obtained to investigate coupling between modules as well as choose control element positions.

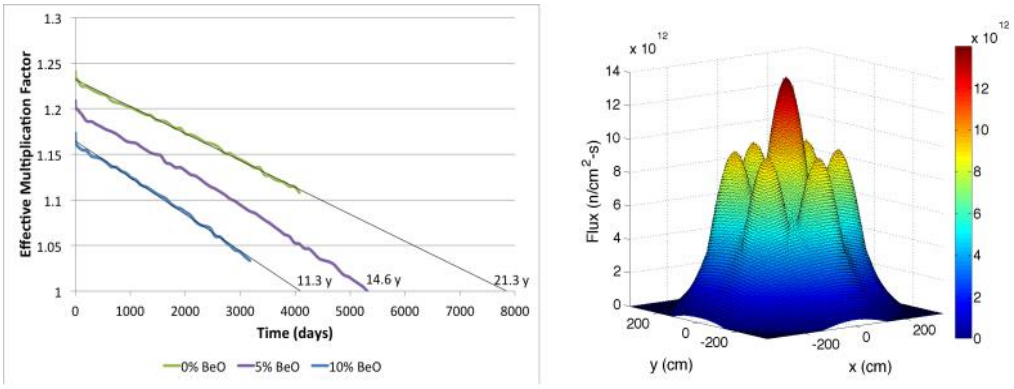

Figure 10. Post processed Serpent results predicting conceptual reactor performance. (Left) The change of the effective multiplication factor as a function of time for various fuel loadings. (Right) Total flux spatial distribution.

Autonomous Corium Reactor [10]

This advanced reactor design (Figure 11) is a fast spectrum system that employ molten metal fuel as the nominal operating state. Models were developed exclusively in Serpent and coupling with STAR-CCM+ to verify operating parameters and transient (thermal and nuclear) behavior.

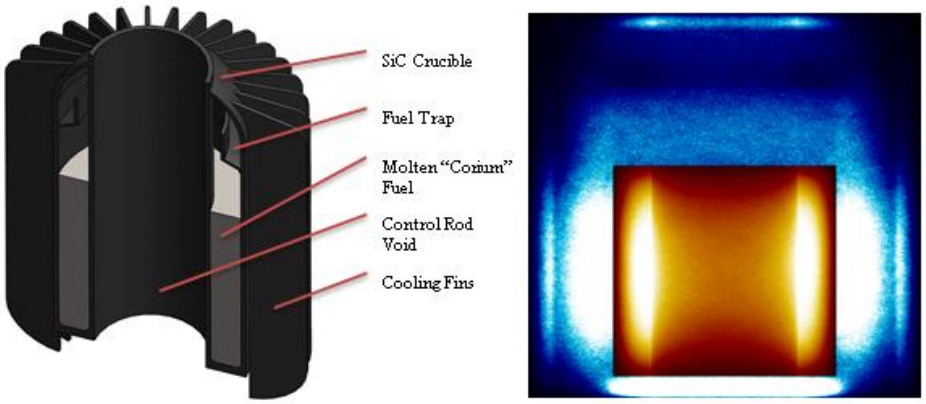

Figure 11. SolidWorks model and Serpent calculation representations for conceptual reactor design.

4.5 SEC NRS

4.5.1 Introduction

The results of Serpent verification for criticality calculations of the uranium-water-graphite systems were presented in the previous Serpent progress report 2010. In this paper results of Serpent 1.1.17 verification for criticality calculations on the basis of a uranium-water critical benchmark (LCT 053, LCT 070, LCT 075, LCT 094 from the International Handbook of Evaluated Criticality Safety Benchmark Experiments [11]) are presented.

4.5.2 Description of experiments

The critical facility consists of core placed in cylindrical tank, which is filled with light water (see figure 12). The criticality of the system is achieved by changing the water level in tank. The core is an arrangement of experimental VVER-type fuel rods at triangular pitch secured between the upper and lower support plate, which are connected to each other by 6 steel columns. The experiments were performed at room temperature.

Number and arrangement of fuel elements in the core as well as fuel enrichment have been varied that have resulted in 43 different core configurations. Summary of all configurations are presented in Table 2.

Table 2. Summary of all configurations.

Experiment	Number of configurations	Number of fuel elements	Lattice pitch, mm	Fuel enrichment, %
LCT-053	14	601 – 2683	12,7	4,4
LCT-070	12	1015 -1201	11	4,4
LCT-075	6	1195	11	4,4
LCT-094	11	1285 – 3853	11	4,4/6,5

Benchmark description also contains results of criticality calculations performed by MCU, Apollo and KENO, so Serpent's accuracy can be compared not only with experimentally measured values but also with values calculated by another codes.

4.5.3 Description of model

The geometry and material composition of all 43 configurations of the critical facility in the model were defined the same way as in the description of the benchmark. The only approximation used in the model is simplification of geometry of fuel rod endings and spring (see figure 12). Thus the model is highly detailed 3D-model of whole critical facility.

All calculations were performed with 500 active cycles and 7000 neutron sources per cycle. Also for better accuracy, unresolved resonance probability tables were used for each calculation. Deviations of calculated Keff values from experimentally measured ones are presented in Tables 3-6. The statistical uncertainty in the calculations is 0.0002. The experimental uncertainties are presented in Tables 3-6 with a confidence interval 3σ.

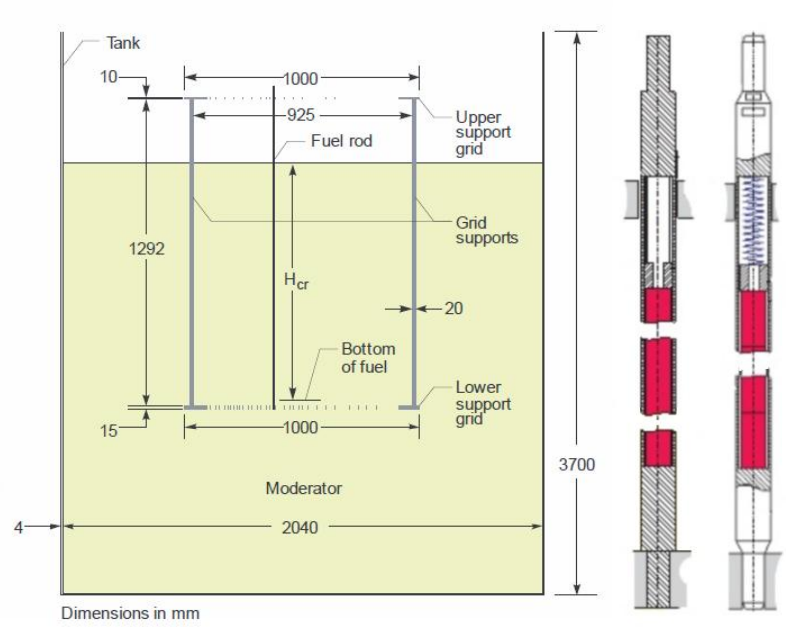

Figure 12. Left: Vertical cross-section of critical facility. Right: Model of fuel rod compared with the actual fuel rod.

Table 3. Deviations of calculated results from experimental data for LCT 053.

Configuration	Serpent	MCU	Experimental uncertainty
1	-0.21%	0.18%	0.48%
2	-0.29%	-0.10%	0.48%
3	-0.14%	-0.01%	0.48%
4	-0.23%	-0.06%	0.48%
5	-0.25%	-0.01%	0.48%
6	-0.32%	-0.05%	0.48%
7	-0.12%	0.10%	0.48%
8	-0.19%	-0.08%	0.48%
9	-0.31%	0.04%	0.48%
10	-0.41%	-0.40%	0.48%
11	-0.24%	-0.02%	0.48%
12	-0.24%	0.07%	0.48%
13	-0.25%	-0.39%	0.48%
14	-0.31%	-0.10%	0.48%

Table 4. Deviations of calculated results from experimental data for LCT 070.

Configuration	Serpent	MCU	Apollo2	KENO	Experimental uncertainty
1	-0.24%	0.25%	0.73%	0.25%	0.51%
2	-0.29%	0.23%	0.76%	0.23%	0.51%
3	-0.24%	0.06%	0.79%	0.06%	0.51%
4	-0.28%	0.21%	0.84%	0.21%	0.51%
5	-0.30%	0.20%	0.94%	0.20%	0.51%
6	-0.35%	0.22%	0.96%	0.22%	0.51%
7	-0.32%	0.19%	0.96%	0.19%	0.51%
8	-0.19%	0.19%	0.97%	0.19%	0.51%
9	-0.41%	0.13%	0.86%	0.13%	0.51%
10	-0.41%	0.16%	0.86%	0.16%	0.51%
11	-0.44%	0.04%	0.74%	0.04%	0.51%
12	-0.24%	0.06%	0.72%	0.06%	0.51%

Table 5. Deviations of calculated results from experimental data for LCT 075.

Configuration	Serpent	MCU	Apollo2	KENO	Experimental uncertainty
1	-0.24%	0.25%	0.73%	0.25%	0.51%
2	-0.29%	0.23%	0.76%	0.23%	0.51%
3	-0.24%	0.06%	0.79%	0.06%	0.51%
4	-0.28%	0.21%	0.84%	0.21%	0.51%
5	-0.30%	0.20%	0.94%	0.20%	0.51%
6	-0.35%	0.22%	0.96%	0.22%	0.51%
7	-0.32%	0.19%	0.96%	0.19%	0.51%
8	-0.19%	0.19%	0.97%	0.19%	0.51%
9	-0.41%	0.13%	0.86%	0.13%	0.51%
10	-0.41%	0.16%	0.86%	0.16%	0.51%
11	-0.44%	0.04%	0.74%	0.04%	0.51%
12	-0.24%	0.06%	0.72%	0.06%	0.51%

Table 6. Deviations of calculated results from experimental data for LCT 094.

Configuration	Serpent	MCU	Apollo2	KENO	Experimental uncertainty
1	-0.24%	0.25%	0.73%	0.25%	0.51%
2	-0.29%	0.23%	0.76%	0.23%	0.51%
3	-0.24%	0.06%	0.79%	0.06%	0.51%
4	-0.28%	0.21%	0.84%	0.21%	0.51%
5	-0.30%	0.20%	0.94%	0.20%	0.51%
6	-0.35%	0.22%	0.96%	0.22%	0.51%
7	-0.32%	0.19%	0.96%	0.19%	0.51%
8	-0.19%	0.19%	0.97%	0.19%	0.51%
9	-0.41%	0.13%	0.86%	0.13%	0.51%
10	-0.41%	0.16%	0.86%	0.16%	0.51%
11	-0.44%	0.04%	0.74%	0.04%	0.51%
12	-0.24%	0.06%	0.72%	0.06%	0.51%

4.5.4 Conclusion

Presented results show a good accuracy of Serpent in case of uranium-water critical systems. The average deviation of Keff values calculated by Serpent from the experimentally measured ones is about 0.25% and the maximum deviation does not exceed 0.43%. The deviation of calculated Keff values for all configurations does not exceed the uncertainty of the experimental values determined with a confidence interval 3σ.

4.6 Électricité de France

The SEPTEN Department of Thermal and Nuclear Studies and Projects is one of the nuclear engineering centers that forms part of EDF's Engineering Production Division. It is the centre of expertise on nuclear safety and for the design of nuclear power stations belonging to EDF.

The SEPTEN establishes the doctrine behind the design of facilities and equipment (principles, rules and technical specifications). It is responsible for demonstrating the safety of these facilities and equipment from their design phase right up until decommissioning.

The SEPTEN is responsible for the consistency of nuclear fuel products and the optimization of their use in reactors.

Lastly, the SEPTEN works on future production solutions and directs EDF's research and development activities in the nuclear energy field.

The SEPTEN boasts a highly skilled and experienced workforce, together with efficient computational and modeling tools in a number of scientific and technical fields: neutronics, reactor physics, thermal hydraulics, mechanics, civil engineering, electrical networks, etc.

- The SEPTEN draws on its skills and resources to provide support to operational power stations in questions relating to nuclear safety in order to improve operational performance and extend operating life to 60 years.
- The SEPTEN supports operational engineering centers with the design certification process and construction of new reactors in France and overseas.
- The SEPTEN oversees the development of new 3rd-generation nuclear reactor models and contributes to preparation for 4th-generation models.

At the SEPTEN, the Reactor Physics Division is involved with fault studies and new nuclear power plant projects. The "Core Software and Physical Models" Group is in charge of the development of the neutronic software used by EDF's Nuclear Production Division. Serpent code has been evaluated in 2011 for its capability to perform Monte Carlo burnup calculations. After a short period of training (due to its very simple and effective syntax), some benchmark have been perform like the Mosteller Benchmark. [12]. Very good agreements were found in comparison with MCNP. Preliminary tests were performing on assembly and 2D

core calculations for the EPR™ reactor (see Figure 13). For 2D core calculations, a new predefined surface type was introduced in the Serpent code. It is called the "gcross" for "generalized cross" in order to define in a easier way the baffle.

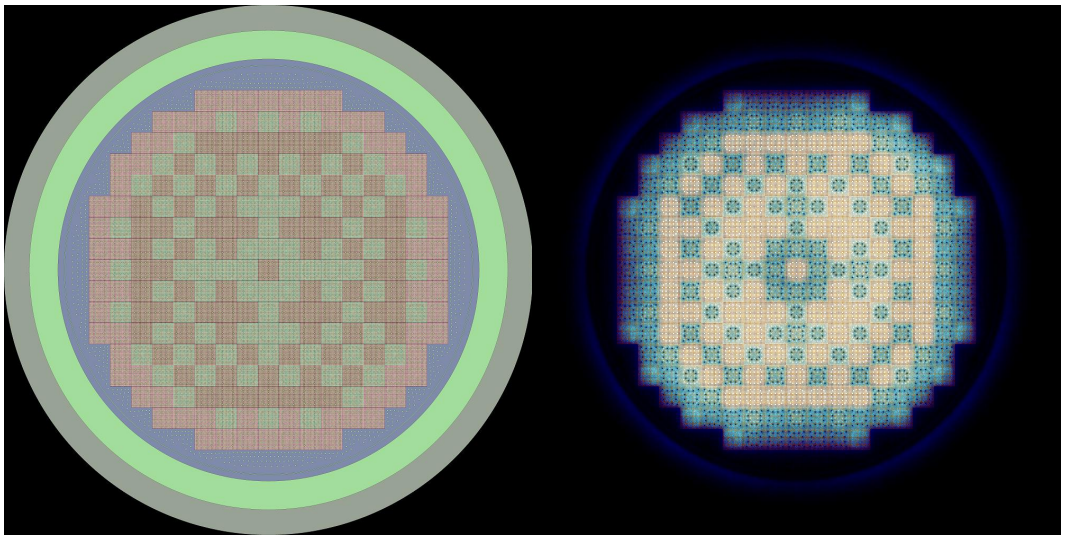

Figure 13. Left: Serpent geometry plot of the EPR core model. Right: Fission rate and thermal flux distribution in the core.

5 Summary, conclusions and future plans

The original motivation for starting the Serpent project about 8 years ago was to extend the use of the continuous-energy Monte Carlo method to tasks traditionally handled using deterministic lattice physics codes. Group constant generation and assembly-level burnup calculations still remain among the main applications. Feedback from the growing user community has lead to a better understanding on what is needed for accomplishing these tasks, in particular the fact that there is still work to be done. One of the main challenges is code validation, which in the future should focus more on comparison to experimental data. A problem shared by the entire reactor physics community is that experimental data, apart from criticality measurements, is relatively scarce, and subject to large uncertainties. This is in particular the case for isotopic measurements from irradiated fuel samples and experiments carried out in power reactor operating conditions.

Serpent development was divided in two parts in late 2010. The new code version, Serpent 2, reached a level of maturity in 2011, and is currently in a beta-testing phase. One reason for re-writing the source code was to remove certain limitations related to parallelization and excessive memory usage in Serpent 1. These goals can now be considered achieved. The second reason is the development of entirely new features and capabilities, which is currently the main focus for the Serpent developer team. Most of the work planned for the near future can be categorized under term "multi-physics", and in practice this implies coupling Serpent externally to thermal hydraulics and fuel performance codes. At VTT the work will additionally involve development of T/H and CFD codes in the framework of a new project funded by the Academy of Finland from October 2012 on. Another new field of application for Serpent 2 is radiation shielding, which involves the development of gamma transport capability and variance reduction techniques.

Inspired by the success of the first International Serpent User Group Meeting in Dresden, Germany, it was decided to arrange another similar meeting in 2012. Madrid, Spain, was selected as the location after a vote between 7 candidates. The meeting will take place 19-21 September, and the number of participants is expected to increase from the previous year.

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