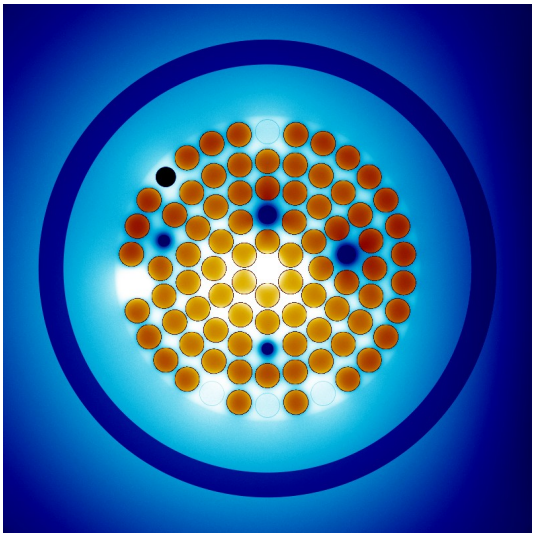
RESEARCH REPORT

VTT-R-01362-11



Serpent Progress Report 2010

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Summary

This report summarizes the work carried out for the development of the Serpent Monte Carlo reactor physics burnup calculation code at VTT Technical Research Centre of Finland during year 2010. New features and major bug fixes are listed and 10 example applications are provided by user organizations.

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Contents

1 Introduction

Serpent is a three-dimensional continuous-energy Monte Carlo neutron transport code, capable of performing burnup calculation. The code is specifically designed for reactor physics applications and the original intended use was the production of homogenized multi-group constants for reactor simulator calculations, similar to conventional deterministic lattice transport codes. However, the versatility of the Monte Carlo method and some additional capabilities have allowed the field of applications to be extended well beyond group constant generation and lattice physics calculations, and Serpent is perhaps best characterized as a Monte Carlo "reactor physics code".

This report summarizes the work related to the Serpent project in the year 2010, although it is not intended to give a full description of the code capabilities. The project is being carried out at VTT Technical Research Centre Finland, with some significant contribution from Finnish universities. Code development is an ongoing effort and feedback from the growing user's community is a valuable and appreciated resource. Some examples of user experience are given in Section 4.

2 Background

The Serpent project 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 (1.0.0) 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 (version 1.1.0) and RSICC in the U.S since March 2010 (version 1.1.7). The NEA base version was later updated to 1.1.7. Some organizations outside the two data centers have acquired the code via bilateral agreements with VTT. The code is licensed for non-commercial research and educational use. Requests for commercial licenses have not yet been made.

A website for the Serpent project was established in October 2008 at http://montecarlo.vtt.fi and it 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 complete list of publications, example input files and a list of recent and up-coming events related to the project. An interactive web-based discussion forum was set up for the user community in March 2010 at http://ttuki.vtt.fi/serpent.

The development work in 2010 was funded from the TOPAS project under the Finnish National Research Program on Nuclear Power Plant Safety, SAFIR2010.

3 Code Development in 2010

The Serpent source code was updated from version 1.1.8 in the beginning of the year to 1.1.14 in November 2010. The main new features and major bug fixes are

described below. More detailed descriptions of each update can be found at the Serpent website.

3.1 New features

Two major improvements were made in the reactor physics capabilities of Serpent in 2010. An equilibrium xenon calculation mode was implemented in update 1.1.9. This method offers a more rigorous approach to the simulation of an operating power reactor, without running a complete burnup calculation.

Production of homogenized few-group constants has been one of the main tasks for Serpent since the beginning of the project. Version 1.1.14 extended this capability to leakage-corrected criticality spectrum calculation, by implementing a separate B_1 fundamental mode solution using micro-group cross sections produced by the Monte Carlo simulation. This method removes some of the incompatibility issues between Serpent and deterministic lattice transport codes, in particular for the calculation of neutron diffusion coefficients. [2]

Even though Serpent is clearly a reactor physics code, it was decided to extend the capabilities to external source calculation for certain special cases where the conventional *k*-eigenvalue criticality source method cannot be used. The implementation was carried out in update 1.1.11, and the source routines are still under development.

In addition to development related to transport methods and reactor physics, some work was also done for the basic nuclear interactions. The Doppler-broadening rejection correction method (DBRC) [3] was implemented in update 1.1.14, and it takes into account the temperature effects on free-gas scattering kernel near low-energy resonances - a physical phenomenon that is completely missed by the standard treatment based on ENDF scattering laws.

3.2 Major bug fixes

The physics routines based on ENDF reaction laws seem to be in a relatively good order, although some minor changes were made in the unresolved resonance probability table treatment and a few bugs were discovered while implementing the external source calculation mode. Most of the major bug fixes were related to data management and parallel calculation mode.

Update 1.1.10 corrected a flaw in the initialization of the random number generator in the MPI mode. Serpent generates pseudo-random numbers using a standard C-function based on a 48 bit linear congruential algorithm, which lacks the capability to reproduce the same random number sequence in single-processor and parallel modes. This, and other topics related to random number generation have been addressed at the discussion forum, and the flaws in the current implementation will be considered in future development.

Serpent stores all information used in the calculation in a single large array of data. The subroutines used 32 bit integer variables to address this data, which caused variable overflow when the array size exceeded 16 GB. To correct this problem, all pointer variables were changed to 64 bit long integers in update 1.1.12.

The combination of parallel calculation mode to unresolved resonance probability table sampling and burnup calculation caused problems that kept re-appearing whenever a new update was issued. These problems were addressed at least in updates 1.1.11 and 1.1.13.

4 User Community

The number of user organizations grew from 20 in January 2010 to 37 by the end of the year. The majority of new users are found in North America, where the RSICC distribution was started in March 2010. There are currently 82 names in the list of registered users, from 20 countries around the world. Serpent discussion forum has been active since March 2010, with 59 members currently signed up.

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 2010 are described below.

4.1 VTT Technical Research Centre of Finland

Most of the work carried out at VTT in 2010 was directly related to code development and validation. Some of this work has been presented in four international conferences and meetings [4-7]. Two scientific review articles were published, one related to the CRAM matrix exponential solver [8], the other describing the delta-tracking based transport routine [9].

A presentation on Serpent and Monte Carlo lattice physics calculations was given at a workshop in the PHYSOR-2010 conference and a similar invited talk at IYNC-2010. Seminars and tutorials were given in four locations (Massachusetts Institute of Technology, University of New Mexico, Studsvik Scandpower and Idaho National Laboratory) during a two-week U.S tour in October 2010.

Methods for performing B_1 fundamental mode calculation for the production of leakage-corrected few-group constants were developed in collaboration with Helmholtz-Zentrum Dresden-Rossendorf (HZDR) [2].

4.2 INVAP, Argentina

Serpent 1.1.7 was used to perform burnup calculations on a theoretical BWR lattice benchmark with burnable poison. The results were compared to benchmark values and reference calculations performed with Condor v.2.61, the INVAP's neutronic collision probability cell code. The comparison involved reactivity and the evolution of U-235, Pu-239 and Gd-155 concentrations as function of fuel burnup. The results showed fairly good agreement between the two codes, which encourages to test Serpent in more complicated geometries as well. The next step will be the generation of few-group constants for MTR type fuel assemblies to be used in full-core calculations. The work carried out at INVAP is published in Reference [10].

4.3 Idaho National Laboratory, USA

Two different organizations at INL are investigating the use of Serpent for future work: The ATR staff and the HTTR staff.

The ATR staff at Idaho National Laboratory is working on upgrading their safety analysis codes for the Advance Test Reactor (ATR). One of the neutronic codes being investigated for modeling the ATR includes Attila. Attila is a powerful 3D discrete ordinate transport code. However, it requires the user to generate its own reactor specific cross sections. Serpent is currently being investigated as one tool to help generate ATR burnup dependent cross sections for use with Attila.

Initial work involved writing an MCNP to Serpent python script that would translate MCNP geometry into the format used by Serpent. The translator was then used to generate a simplified model of the ATR in Serpent, seen in Figure 1. Future work include running Serpent on a full detailed model of the ATR, generating burnup dependent cross sections for Attila, and comparing Attila results using those cross sections to results from cross sections generated by SCALE6.

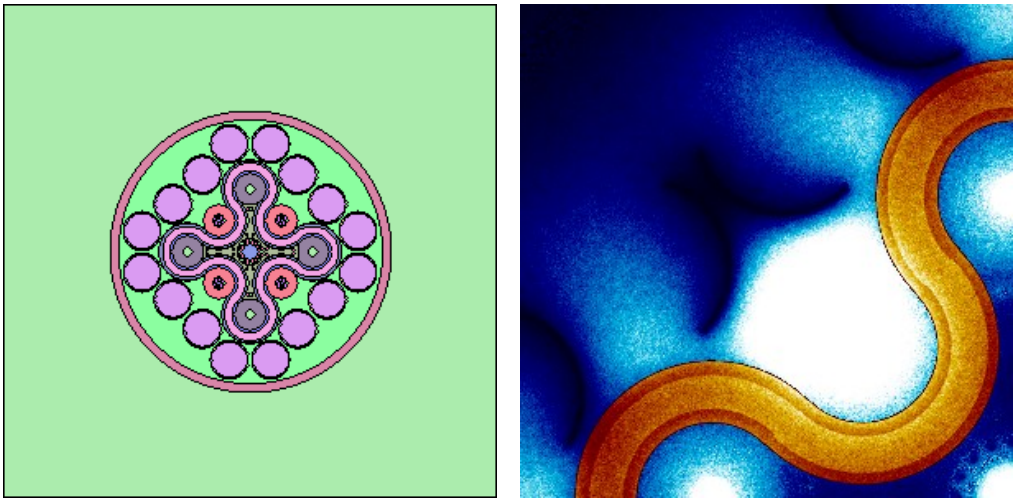

Figure 1. Left: Serpent geometry plot of the simplified ATR model. Right: Thermal flux and fission rate distribution in an ATR core quadrant.

The HTTR staff at INL has been using Serpent for benchmarking purposes of a deterministic model of the high temperature test reactor (see the Figure 2). The advantage of Serpent is its capability to generate diffusion parameters including coefficients and full scattering matrixes. A 26-group data cross section set was generated with Serpent and compared to cross sections from DRAGON 4. The calculated eigenvalue from the cross sections generated with Serpent was almost 4000 pcm lower than the reference MC calculation. Whereas the cross sections from DRAGON 4 were 1793 pcm above the reference solution. Future work with Serpent includes investigating further the Serpent cross section generation capability for the HTRs by comparing results from Serpent to other lattice codes like HELIOS and SCALE6. Two papers, published by the HTGR staff at INL, contain a detailed description the work performed with Serpent [11,12].

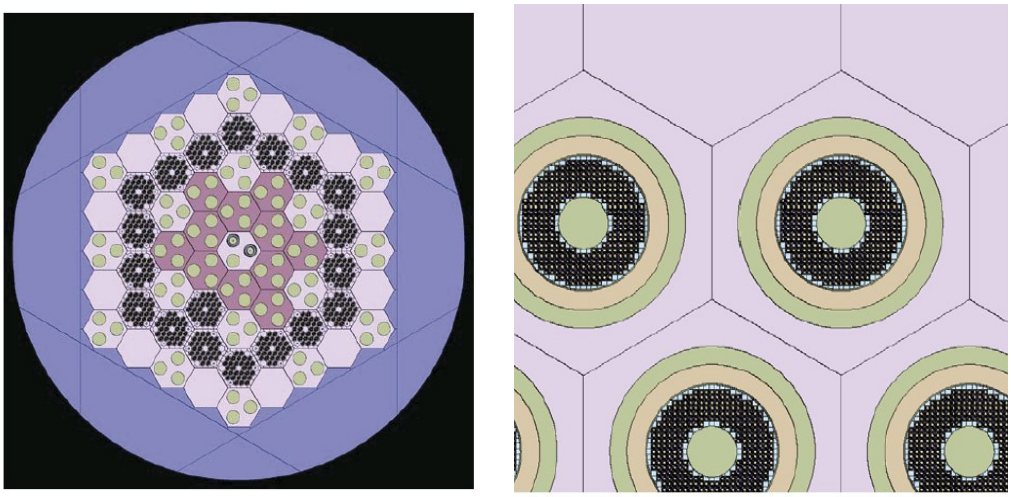

Figure 2. Left: Hex-plane view of the HTTR Serpent model. Right: TRISO detail within the fuel compact in the HTTR Serpent model.

4.4 Colorado School of Mines, USA

At the Colorado School of Mines Serpent, in conjunction with MCNPX2.6.0, assesses the feasibility of improving reactor core burnup performance of a deep burnup prismatic Very-High Temperature Reactor. In this project, modifying the tri-structural-isotropic fuel particles to add a layer of burnable poison on the outside of the uranium carboxide (UCO1.5) fuel kernel to allow a higher initial fuel loading and reduce the reactor's operating reactivity swing. The project will use Serpent to verify the MCNPX2.6.0 results as the project evolves.

One part of the Fall 2010 reactor design class used the Serpent code to provide verification of MCNP and SCALE. The class focused on the neutronicsnics assessment of two different reactor designs, and Serpent calculated keff, fluxes within the core, and burnup.nup. The design teams compared the findings from the Serpent code to nearly identical models created in MCNP and SCALE. The Final design reports for each team contained the results and comparisons.

4.5 Lappeenranta University of Technology, Finland

Serpent was used for researching pebble-bed type high temperature gas cooled reactors in the Laboratory of Nuclear Engineering during the year 2010. Five different benchmark configurations of ASTRA criticality facility were modeled with Serpent. ASTRA benchmark has been done by Russian Scientific Centre Kurchatov Institute and description about these experiments is available from OECD NEA. Multiplication factors were calculated for each configuration and results reported in HTR 2010 conference [13].

Other application for the code has been power distribution calculations in pebble bed reactors. These calculations are part of coupled calculations of reactor physics and thermal hydraulics. Randomly packed pebble beds are generated by using a discrete element method (DEM) code and the resulting configurations are used in Serpent. Serpent will calculate the power distribution in the core and give the power per pebble as a result. The power distribution is then mapped in a

computational fluid dynamics (CFD) mesh and the thermal hydraulic behavior of reactor is solved.

Serpent has been used to generate group constants for a student exercise and there are plans for using Serpent more widely in the teaching of reactor physics in our courses.

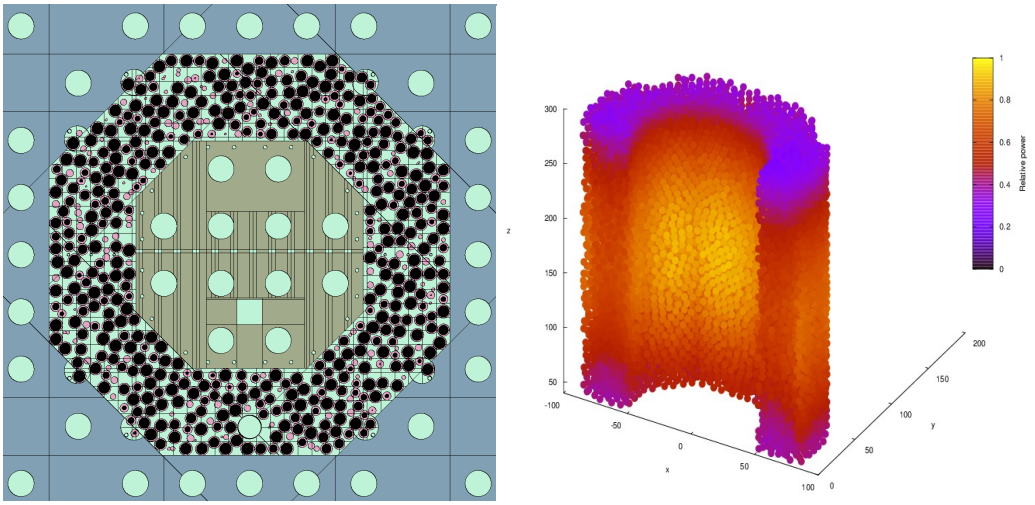

Figure 3. Left: Cross-sectional view of the Serpent model of the ASTRA critical facility. Right: Example of power distribution of the randomly packed pebble bed.

4.6 Massachusetts Institute of Technology (MIT), USA

Serpent is being used at MIT to model the neutronics of Advanced Light Water Reactors that have the potential to breed and consume transuranic actinides to achieve a high conversion ratio [14]. Modeling the neutronics of this type of reactors is difficult because of the very axially heterogeneous nature of these designs with fissile zones producing neutrons and blanket zones consuming them. A typical design and relative assembly axial power distribution as obtained from Serpent results is shown in Figure 4.

Conventional methods fail to reflect correctly the effects of spectral overlap between the neighboring heterogeneous assembly zones because each material zone is decoupled from other zones. In these new designs, because the spectrum is continuously changing along the axial direction due to increasing void fraction, a new strategy must be employed to generate cross sections. Since Serpent is a Monte Carlo code and allows for the homogenization of many zones at once, few-group homogenized parameters can be generated for each zone at the same time. Therefore, zones do not have to be decoupled from one another and homogenized cross sections can be calculated in the presence of other zones where axial streaming of neutrons can be captured.

To efficiently generate cross sections, a Python wrapper has been developed to generate branch cases of cross sections based on perturbations of operating conditions. The wrapper, Serpent XS, has been developed for use with the U.S. NRC Core Simulator PARCS [15]. The user can run her Serpent input file with slight modifications along with a branch case input file to generate a PARCS

PMAXS cross section database file. PARCS can then use this file for interpolation for steady state and transient analyses. Initial results confirm implementation of Serpent-PARCS linking procedure was performed correctly [16]. Figure 5 shows an example of a fuel temperature branch case comparing Serpent and PARCS. This tool will allow us to analyze the impact of generating cross sections with and without influence from other zones and how these differences will impact the full core reactor analysis.

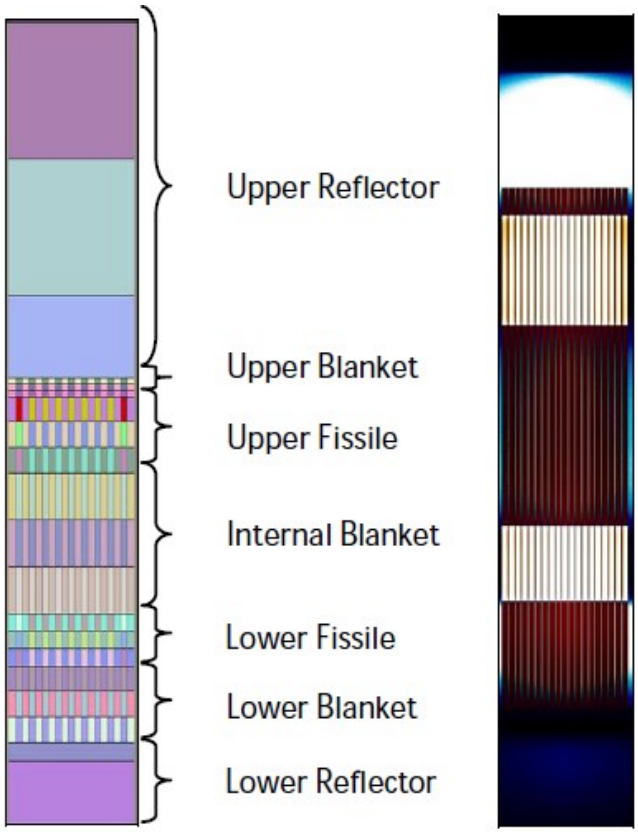

Figure 4. Left: Axial view of center of assembly. Right: Axial power distribution from Serpent.

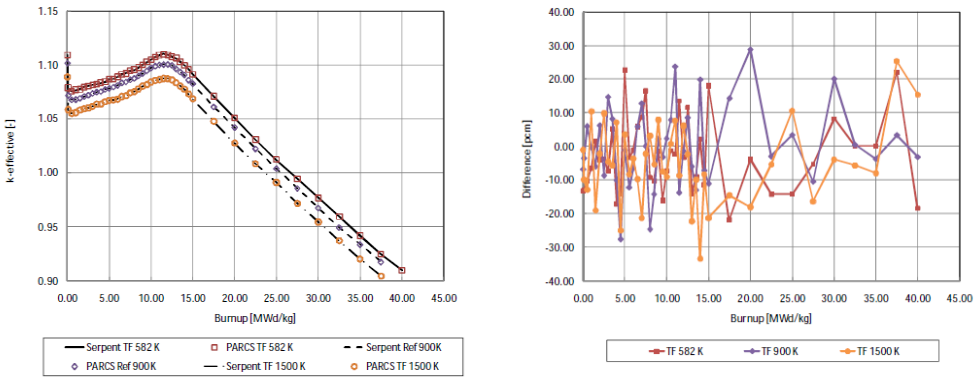

Figure 5. Fuel temperature branch (TF) comparison of k-effective between Serpent and PARCS using cross sections generated from Serpent for a typical PWR with Gadolinium pins. Left: k-effective vs. burnup. Right: difference in k-effective.

4.7 École Polytechnique de Montréal, Canada

The Serpent Monte Carlo code is intensively used at the Nuclear Engineering Institute of the Ecole Polytechnique of Montreal since summer 09. It serves in some M.Sc. or Ph.D. projects as well as in a graduate course on computational reactor physics.

M.Sc. and Ph.D projects

Up to now, the most common usage of the Serpent code at EPM is to serve as a reference tool to assess some recent developments done in the lattice code DRAGON. In a project intended to extend the method of Collision Probability and the method of Characteristics to 3D spherical geometries, some comparisons have been made between DRAGON and Serpent and simple cases. The objective was eventually to simulate the neutronic behavior of TRISO particles used in Very high Temperature Reactors. These fuel elements present both the properties of being spherical and randomly distributed in the medium, which cannot be analyzed by traditional deterministic lattice codes, hence the use of Serpent. Random positions of the elements were generated in DRAGON, and then fed to Serpent, and analyzed by both codes, and both reaction rates and fluxes were compared, with excellent concordance between the two.

Serpent was also used in code-to-code comparisons with a Monte Carlo lattice code based on cross-section probability tables developed in 2010. In this case, reaction rates on simple PWR pin cells and assemblies were compared. Comparative studies were performed to assess the probability table method compared to continuous-energy Monte Carlo reference calculation. Group constants produced by both codes were also compared. The recent implementation of a leakage model based on the B1 homogeneous fundamental mode in Serpent version 1.14 was very helpful. It was typically confronted to (1) the DRAGON lattice code and (2) the mentioned Monte Carlo with probability table lattice code, in which a B1 fundamental mode calculation was implemented as well. Results were found very consistent between all the codes. Two journal papers with Serpent/DRAGON code-to-code comparisons are published [17,18].

Graduate course on reactor physics

Serpent serves to illustrate Monte Carlo-based reactor physics calculations in a graduate course on computational reactor physics. At least one assignment is based on the Serpent code. For instance in 2010, it was on the evaluation of the Doppler reactivity effect for a PWR pin cell (the “Mosteller” benchmarks).

4.8 SEC NRS, Russia

Serpent 1.1.13 was validated by performing criticality calculations on the basis of a uranium-graphite critical benchmark (evaluation LEU-COMP-THERM-060 in the International Handbook of Evaluated Criticality Safety Benchmark Experiments).

The graphite stack of the RBMK critical facility consists of 324 (18x18) channels. The experiments were performed at room temperature. The maximum power of the RBMK critical facility is 25 watts. All elements of the facility core are

identical to similar elements of the RBMK reactor except the height of the core, which is half that of the power reactor. The critical masses studied were in the form of uniform configurations, configurations with empty channels in the core, with water columns, with additional standard absorbers made of boron steel, and a few with thorium absorbers [19].

Calculation results

The geometry and material composition of all 28 experiments were defined in a same way as it was defined in the description of the benchmark's experiment – that is highly detailed 3D-model of whole critical facility. So in that case we can estimate Serpent accuracy by comparing of the Serpent neutron multiplication factor calculation results with experimental and calculated by another codes values. At figure 6 part of the fuel assembly model are represented.

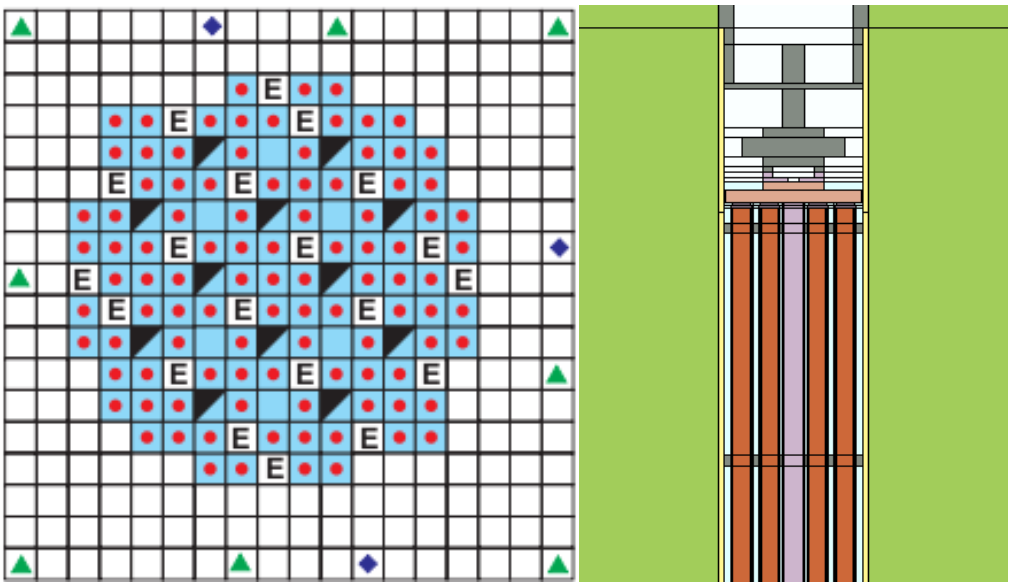

Figure 6. Left: cartogram for experiment #17. Right: Model of fuel assembly.

The following parameters were used for calculations:

- 2000 neutron sources per cycle;
- 500 active cycles.

Also for better accuracy, unresolved resonance probability tables were used for each experiment. Calculations of k_{eff} results are presented in table 1 (in the brackets deviations of calculated k_{eff} with experimental values are indicated).

The statistical uncertainties are 0.0002 in case by MCNP 4C [20] calculations and 0.0006 in case of Serpent calculations. In case of MCU [21] calculations statistical uncertainties are not presented in the benchmark's description.

Table 1. Calculated and experiment k_{eff}.

№	Serpent 1.1.13 (ENDF/B-6.8)	MCNP 4C (ENDF/B-V)	MCU (DLC/MCUDAT-2.1)	Experiment
1	1.0003 (0,13%)	1.0023 (-0,33%)	1.0024 (-0,34%)	0.9990
2	0.9975 (-0,02%)	1.0020 (-0,43%)	1.0069 (-0,92%)	0.9977
3	1.0023 (0,22%)	1.0031 (-0,30%)	1.0049 (-0,48%)	1.0001
4	1.0023 (0,06%)	1.0044 (-0,27%)	1.0087 (-0,70%)	1.0017
5	1.0043 (0,26%)	1.0043 (-0,26%)	1.0049 (-0,32%)	1.0017
6	0.9899 (-0,03%)	0.9937 (-0,35%)	0.9982 (-0,81%)	0.9902
7	1.0070 (0,42%)	1.0075 (-0,47%)	1.0085 (-0,57%)	1.0028
8	1.0045 (0,06%)	1.0106 (-0,67%)	1.0149 (-1,10%)	1.0039
9	1.0071 (0,28%)	1.0082 (-0,39%)	1.0079 (-0,36%)	1.0043
10	1.0010 (-0,04%)	1.0061 (-0,33%)	1.0085 (-0,34%)	1.0014
11	1.0049 (0,48%)	1.0043 (-0,47%)	1.0062 (-0,71%)	1.0001
12	1.0046 (0,37%)	1.0038 (-0,42%)	1.0039 (-0,61%)	1.0009
13	1.0034 (0,24%)	1.0038 (-0,29%)	1.0036 (-0,30%)	1.0010
14	0.9984 (-0,31%)	1.0036 (-0,28%)	1.0083 (-0,26%)	1.0015
15	1.0015 (0,03%)	1.0017 (-0,21%)	1.0022 (-0,68%)	1.0012
16	0.9980 (-0,27%)	0.9989 (-0,05%)	1.0037 (-0,10%)	1.0007
17	1.0053 (0,38%)	1.0058 (0,18%)	1.0071 (-0,30%)	1.0015
18	1.0001 (-0,08%)	1.0054 (-0,43%)	1.0086 (-0,56%)	1.0009
19	1.0030 (0,20%)	1.0049 (-0,45%)	1.0044 (-0,77%)	1.0010
20	1.0032 (0,10%)	1.0094 (-0,39%)	1.0061 (-0,34%)	1.0042
21	1.0034 (0,28%)	1.0043 (-0,52%)	1.0044 (-0,19%)	1.0006
22	1.0038 (-0,01%)	1.0084 (-0,37%)	1.0051 (-0,38%)	1.0039
23	1.0036 (0,16%)	1.0057 (-0,45%)	1.0064 (-0,12%)	1.0020
24	1.0013 (-0,15%)	1.0070 (-0,37%)	1.0043 (-0,44%)	1.0028
25	1.0051 (0,35%)	1.0060 (-0,42%)	1.0062 (-0,15%)	1.0016
26	0.9998 (0,02%)	1.0052 (-0,44%)	1.0061 (-0,46%)	0.9996
27	1.0056 (0,25%)	1.0061 (-0,56%)	1.0050 (-0,65%)	1.0031
28	1.0016 (-0,12%)	1.0063 (-0,30%)	1.0083 (-0,19%)	1.0028

Conclusion

Presented results showed a good accuracy of Serpent in case of uranium-graphite critical systems. The deviations are low enough and in the most of experiments even lower than in cases of MCNP 4C and MCU.

4.9 Atomic Energy of Canada Limited (AECL)

Solving the 3-dimensional neutron-diffusion equation with 2-group lattice-averaged cross sections has been traditionally used for the design and safety analysis of the CANDU and LWR full cores. To assess the adequacy of such method for the prediction of the core k-effective and core-wide channel-average power distribution, benchmark problems were set up with the aid of the Serpent code for CANDU and PWR respectively. Figure 7 illustrates the geometry of the CANDU benchmark problem.

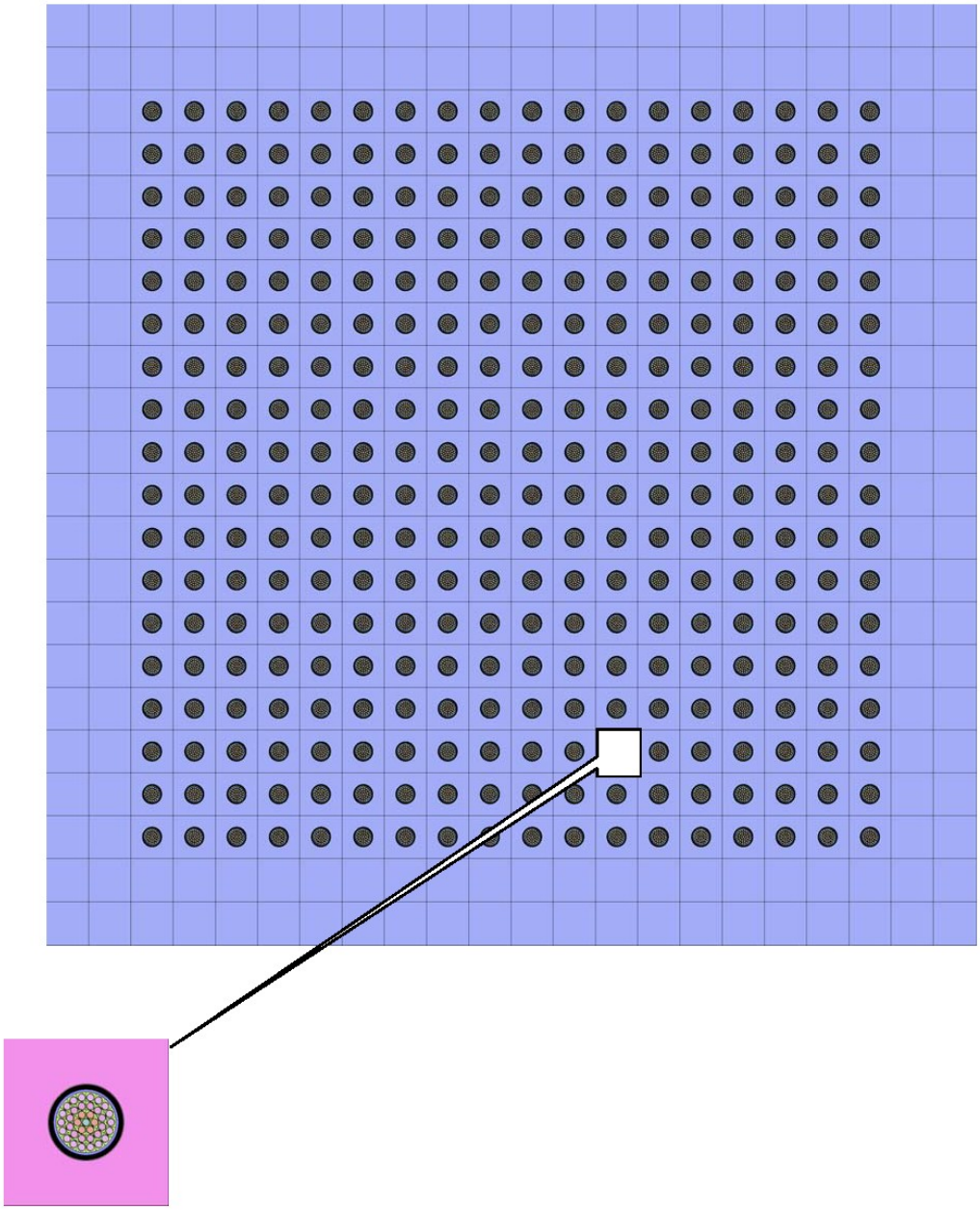

Figure 7. Illustration of the CANDU benchmark problem.

Serpent is used in the assessment for the following two objectives: one is to provide reference flux/power distributions for the benchmark problems; the other is to provide lattice-homogenized cross sections based on the Serpent transport calculations.

Since it's the first time for us to use Serpent at AECL, our first task is to benchmark Serpent calculations against the reference MCNP results for the 2-D CANDU benchmark problem, to build up confidence on Serpent and our use of Serpent. To obtain statistically converged flux/power distributions and lattice-homogenized cross sections for the benchmark problems, various histories were tested in the Serpent calculations for the CANDU benchmark problem with the exact same nuclear data library as that of MCNP5.

The difference in the core k-effective value between Serpent and MCNP5 is about 30 pcm (0.3 mk). Figure 8 shows the differences in channel powers for the whole core between Serpent (with 800 million histories) and MCNP5 (with 2300 million histories) for the 2-D CANDU benchmark problem. The RMS difference in the channel power between Serpent and MCNP is about 0.2%. Overall the assessment shows that Serpent's prediction of core k-effective and channel powers is very close to those predicted with MCNP5.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
A	0.2	0.3	0.3	0.2	0.1	0.1	0.0	-0.1	-0.1	-0.2	-0.2	0.0	0.0	0.1	0.0	0.0	-0.2	-0.1
B	0.3	0.2	0.0	-0.1	0.2	0.0	0.0	0.0	-0.2	-0.1	0.0	-0.2	0.0	0.0	-0.1	-0.2	-0.1	-0.1
C	0.2	0.1	0.0	0.0	0.1	0.0	0.0	-0.1	-0.1	-0.1	0.0	-0.1	-0.1	-0.1	-0.2	0.0	-0.1	0.0
D	0.2	0.3	0.1	0.1	0.0	0.0	0.0	-0.1	-0.2	-0.1	0.0	0.0	0.0	-0.2	-0.1	-0.1	0.0	-0.3
E	0.3	0.2	0.1	0.1	0.0	0.0	-0.1	-0.1	-0.1	0.0	-0.1	0.0	-0.1	-0.2	-0.1	-0.1	0.0	-0.1
F	0.5	0.5	0.4	0.2	0.2	0.1	0.1	0.0	0.0	0.0	0.0	0.0	-0.1	-0.1	-0.1	-0.1	0.0	-0.1
G	0.4	0.4	0.3	0.3	0.3	0.3	0.2	0.1	0.1	0.0	0.1	0.0	0.0	0.0	-0.1	-0.1	-0.2	-0.1
H	0.4	0.5	0.4	0.3	0.3	0.2	0.2	0.0	0.0	0.0	-0.1	0.0	-0.1	0.0	0.0	0.0	0.0	-0.2
J	0.3	0.4	0.5	0.5	0.4	0.3	0.1	0.0	0.0	0.1	0.0	-0.1	-0.1	0.0	0.0	-0.1	-0.1	-0.1
K	0.3	0.2	0.4	0.5	0.4	0.3	0.1	0.0	-0.1	0.1	0.0	0.0	-0.1	0.0	0.0	0.0	-0.1	0.1
L	0.2	0.4	0.4	0.3	0.4	0.2	0.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	-0.1
M	0.2	0.2	0.2	0.1	0.2	0.2	0.0	0.1	0.1	0.0	0.0	0.1	0.0	0.0	0.1	0.0	-0.1	-0.1
N	0.1	0.3	0.2	0.1	0.0	0.0	-0.1	0.0	0.0	0.0	-0.1	0.0	-0.2	0.0	-0.1	-0.1	-0.2	-0.2
O	0.0	0.0	0.0	0.0	-0.1	-0.1	-0.1	-0.2	-0.1	0.0	-0.1	-0.2	-0.2	-0.1	-0.2	-0.2	-0.3	-0.2
P	-0.1	0.0	-0.1	0.0	-0.1	0.0	-0.1	-0.2	-0.1	-0.2	0.0	0.0	-0.2	-0.2	-0.2	-0.2	-0.1	-0.4
Q	-0.1	-0.1	-0.2	-0.2	-0.1	0.0	-0.2	-0.2	-0.2	-0.2	-0.1	-0.3	-0.3	-0.3	-0.3	-0.4	-0.3	-0.3
R	0.1	0.0	-0.1	-0.1	0.0	0.0	-0.1	-0.2	-0.4	-0.3	-0.2	-0.3	-0.3	-0.2	-0.5	-0.5	-0.3	-0.2
S	-0.1	-0.2	0.0	-0.1	-0.1	-0.2	-0.1	-0.3	-0.2	-0.3	-0.2	-0.1	-0.4	-0.4	-0.2	-0.1	-0.4	0.0

Figure 8. Differences (%) in Channel Powers Between Serpent and MCNP Calculations for the CANDU Benchmark Problem.

Most of Serpent calculations were executed with Serpent 1.1.12 with a single CPU on the LINUX platform at AECL in 2011. Now most Serpent calculations were executed with Serpent 1.1.14 with the parallel mode. Compared with other Monte Carlo codes, Serpent is powerful (depletion, cross-section generation, flux/power distribution), efficient (about 10-40 times faster than MCNP), and user friendly. As a third-party tool, Serpent assists us in performing the assessment work without surprises. In the meantime, some deficiencies (such as huge memory requirement, unreliable diffusion coefficients, etc.) in Serpent have also been uncovered and reported to the code developer.

4.10 Aalto University, Finland

In year 2010, the fission research group of Aalto University (former Helsinki University of Technology) utilized Serpent in a wide variety of purposes.

It was used as a testing platform when developing methods for burnup calculations with Monte Carlo neutronics and full system of nuclides [22,23]. Significant improvements to the burnup algorithms were discovered and are planned to be implemented to future versions of Serpent.

Serpent was successfully combined with an external temperature distribution solver for a cylindrical fuel rod situated in a pin cell. In this configuration, Serpent is used to resolve the power and burnup distributions within a fuel rod that is divided radially into numerous rings to reach proper spatial resolution. The temperature distribution can be further returned to Serpent and, hence, the code combination is able to simulate also the temperature feedback on neutronics.

Results of this code system were utilized mostly to study the concept of effective temperatures in a fuel rod.

Serpent was also used to model a few larger systems. The FiR-1 reactor (Triga Mk II) was modeled and the model was used to assess the current burnup distribution and nuclide inventory of the reactor. However, the extreme complexity of the research reactor's power history was quite challenging to approximate in the burnup calculation, which caused some uncertainty in the results. Serpent was also compared to a deterministic code CASMO-4E in the case of a Radkowsky Seed Blanket Unit, which contains thorium. The results of the codes proved to be quite similar, but Serpent predicted the conversion of Thorium to Uranium 233 to be slightly higher than CASMO-4E.

5 Summary, Conclusions and Future Plans

The Serpent user community grew significantly during 2010, especially in North America where the RSICC distribution was started in March. The code is mainly used for three purposes: group constant generation, validation of deterministic transport codes and research reactor modeling. User feedback has been mostly positive, and extremely valuable for the development work.

Code development during the past six months has mainly focused on minor improvements and special capabilities required for some particular task. The source code seems to be in a relatively stable state, and apart from a few fundamental flaws discussed below, there are no known major bugs or problems in the calculation routines.

The major development task for the near future will be re-writing the entire source code. The reason for this is that the basic structure of the program has become so complicated that it is very difficult to develop new capabilities while keeping everything together. This is especially the case for certain routines related to data management and parallelization, which eventually results in memory usage becoming a limiting factor in burnup calculation.

The work on Serpent 2, or “Super-Serpent”, has already begun, and a related discussion area has been opened at the user forum. Work on Serpent 1 still continues, but all major development goals will be deferred to the next version. All main functionality in the current code version will be preserved and additional plans include:

- Parallelization based on the combination of distributed (MPI) and shared memory (OpenMP) techniques
- Capability to handle up to 100,000 material regions in burnup calculation in full parallel mode
- Combined neutron/gamma transport simulations
- On-the-fly Doppler-broadening for reaction cross sections

The first international Serpent user group meeting and workshop will be held in Dresden, Germany, 15.-16. September 2011. The intention is to bring Serpent

users together to present their work and discuss various topics related to the code and its applications, as well as the future development of version 2.

The main source of funding for Serpent development, the Finnish National Research Program on Nuclear Power Plant Safety (SAFIR2010) has come to an end. The work will continue in the new SAFIR2014 program, with a new project (KÄÄRME) allocated for Serpent development. The funding covers 16 person-months of work in 2011, and an additional funding application to the Academy of Finland is currently pending.

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