Serpent Progress Report 2012

Authors: Jaakko Leppänen
Confidentiality: Public
This report summarizes the work carried out for the development of the Serpent Monte Carlo code during year 2012.
Contents

Contents.................................................................................................................................................. 2

1. Introduction......................................................................................................................................... 3

2. Background......................................................................................................................................... 3

3. Code development in 2012 ................................................................................................................. 3

4. User Community ............................................................................................................................... 4
   4.1 Royal Military College of Canada (RMCC)................................................................................... 4
   4.2 GeNERG-DIME/TEC - University of Genova ........................................................................... 8
      4.2.1 Introduction and model presentation .................................................................................. 8
      4.2.2 Results .............................................................................................................................. 11
   4.3 SECNRS ...................................................................................................................................... 13
      4.3.1 Introduction ...................................................................................................................... 13
      4.3.2 Description of experiments ............................................................................................. 14
      4.3.3 Description of model ........................................................................................................ 17
      4.3.4 Results .............................................................................................................................. 18
      4.3.5 Conclusion ......................................................................................................................... 20
   4.4 Électricité de France ..................................................................................................................... 20

5. Summary, conclusions and future plans............................................................................................ 23

Acknowledgements ............................................................................................................................... 24

References .............................................................................................................................................. 24
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 was originally specialized in assembly-level calculations, including depletion studies and the production of homogenized multi-group constants for full-core reactor simulator calculations, but the field of applications has been considerably broadened over the years. The on-going work is mainly focused on a new code version, Serpent 2, with a significant emphasis in coupled multi-physics calculations.

This report summarizes the work carried out for the development of Serpent 1 and Serpent 2 in 2012 and early 2013. 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 pre-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). In the on-going SAFIR 2014 program, Serpent development is carried out under the KÄÄRME project, specifically established for the task. Additional funding is received from the EU High Performance Monte Carlo Reactor Core Analysis (HPMC) project, and the Academy of Finland Numerical Multi Physics (NUMPS) project.

3. Code development in 2012

Serpent 1 source code was updated from version 1.1.16 at the beginning of the year to 1.1.19 in April 2013. Update 1.1.17 covered several features suggested by users during the first international Serpent User Group meeting (see discussion notes at http://montecarlo.vtt.fi/mtg/2011_Dresden/index.htm and Sepent Progress report 2011 [2]). The other two updates are mainly related to bug fixes and minor revisions in existing features.
The main focus in the development of new features has moved from Serpent 1 to Serpent 2, which is currently available to registered users by request. The public distribution of Serpent 2 is scheduled for late 2014. The work is currently focused on two specific topics:

1) Advanced methods for spatial homogenization
2) Coupled multi-physics applications

Work on surface current tallies and B1 leakage correction to homogenized group constants, started in 2011, was completed in 2012 [3,4]. Several subroutines related to group constant generation were re-written, and new input features added for the automated calculation of assembly discontinuity factors (ADFs) and spatial form functions for pin-power reconstruction. Work was also started for the calculation of homogeneous diffusion flux solution for obtaining ADFs in geometries homogenized without zero net current boundary conditions [5].

Development of a multi-physics coupling scheme was started in late 2012 [6], when additional funding was received for the purpose from the Academy of Finland NUMPS project. The work was divided between the development of internal solvers for thermal hydraulics and fuel behavior [7,8], and external coupling via a universal multi-physics interface [9,10]. Development of an on-the-fly temperature treatment routine based on the target motion sampling (TMS) method was continued [11-13], and the external source simulation mode extended to cover time dependence of neutronics [14].

Work on the development of the CRAM matrix exponential method was summarized in a doctoral thesis, completed in 2013 [15].

4. User Community

The number of user organizations grew from 57 at the end of 2011 to 88 in January 2013. The number of registered users increased to about 220. The Second International Serpent User Group Meeting was hosted by the Universidad Politécnica de Madrid in Madrid, Spain, on September 19-21, 2012. The meeting brought together 40 Serpent users from 16 Organizations in Europe and the U.S. The presentations are available for downloading at website: http://montecarlo.vtt.fi/mtg/2012_Madrid/index.htm.

The applications for the Serpent code have ranged from group constant generation and fuel cycle studies to research reactor calculations and coupled multi-physics applications. Some example user applications are described below.

4.1 Royal Military College of Canada (RMCC)

Researchers at the Department of Chemistry and Chemical Engineering, Royal Military College of Canada (RMCC), have been verifying the capabilities of Serpent 1.1.19 for the multispectrum CANDU reactor calculations [16]. As preliminary calculation, the two-region test reactor known as the Deuterium Critical Assembly, DCA (Figure 1) is simulated by both MCNP5 and Serpent 1.1.19. The verification was performed by calculating the multiplication factor \( k_{\text{eff}} \), criticality factors of each region \( k_{11} \) and \( k_{22} \) and coupling coefficients \( k_{12} \) and \( k_{21} \). The fluxes of three neutron energy groups (Fast, epithermal and thermal) were calculated using both MCNP5 and Serpent.

The multiplication factors \( k_{\text{eff}} \) that calculated numerically and independently from simulations of the DCA by MCNP5 and Serpent codes were compared with the multiplication factors \( k_{\text{eff}} \) calculated based on the coupled reactor theory. Excellent agreement was obtained between
the multiplication factors $k_{\text{eff}}$ calculated with the Serpent code, with MCNP5, and that from the coupled reactor theory. A very good agreement of fluxes calculations for three energy neutron’s groups (Fast, epithermal and thermal), were found between the two codes calculations. A very good agreement was obtained between $k_{\text{eff}}$ numerically calculated from Serpent and those from the MCNP5. Therefore, the coupling coefficients can be calculated with sufficient accuracy using these codes. This analysis demonstrates that the Serpent code is valid for the multipoint coupled reactor calculations.

1 - From Figure 2, one can notice the consistency of the criticality factors $k_{11}$, $k_{22}$ and coupling coefficients $k_{12}$ and $k_{21}$ as calculated with MCNP5 and Serpent. Consequently, the multiplication factors $k_{\text{eff}}$ as calculated from the coupled reactor theory.

2 - Figure 3 represents the comparison between the multiplication factors $k_{\text{eff}}$ calculated directly by both MCNP5 and Serpent and that calculated by the coupled reactor theory. One can notice that the behaviours of the $k_{\text{eff}}$ curves calculated by the two codes are very similar. This is also true for the $k_{\text{eff}}$ calculated from the coupled reactor theory.

3 - Due to the limitation of Serpent for the flagging cell flux detector definition and because of the strong agreement between the fluxes in fast core and thermal core, $\phi_1(E_i)$ and $\phi_2(E_i)$, calculated by MCNP5 and those calculated by Serpent. the flagged waiting flux from MCNP5 could be used to calculate the flagging flux in Serpent. These flux values are normalized per number of neutrons history and volume. The flagging flux from the thermal core in the air gap cell 5 in Serpent, $\phi_1(E_{i,1-2})(\text{serpent})$, could be calculated by multiplying the thermal flux of core $\phi_2(E_i)$ from Serpent times the ratio between the flagging flux $\phi_1(E_{i,1-2})(\text{MCNP})$ and the thermal flux of core $\phi_2(E_i)(\text{MCNP})$. This ratio is called flagged waiting flux as shown in Equations (1). The same method is used to calculate the flagging flux from the fast core to the thermal one:

$$\phi_1(E_{i,1-2})(\text{Serpent}) = \frac{\phi_1(E_{i,1-2})(\text{MCNP})}{\phi_1(E_{i})(\text{MCNP})} \times \phi_1(E_{i})(\text{Serpent})$$

4 - Table 1 presents the average percentage difference between the $k_{\text{eff}}$ calculated by MCNP5 and Serpent and from the coupled reactor theory.

5 - In this part of the simulation, the level of light water in the fast core was fixed at 100 cm while the level of the heavy water was changed from 0 to 200 cm as shown in Figures 4 and Figure 5. One can notice the consistency of the criticality factors $k_{11}$, $k_{22}$ and coupling coefficients $k_{12}$ and $k_{21}$ as calculated from the MCNP5 and Serpent codes. Consequently, The multiplication factors $k_{\text{eff}}$ calculated by the coupled reactor theory, depending on these values $k_{11}$, $k_{22}$, $k_{12}$ and $k_{21}$, from the two codes independently, are almost match the $k_{\text{eff}}$ calculated with MCNP5 and Serpent independently.

6 - Figure 4 presents the comparison between the multiplication factors $k_{\text{eff}}$ values calculated directly by both Serpent and MCNP5 and those calculated with the coupled reactor theory. One can notice that the behaviour of $k_{\text{eff}}$ curves calculated by the two codes corresponds well and are also close to the values determined with the coupled reactor theory. Table (1) describes the average percentage difference between the $k_{\text{eff}}$ calculated by MCNP5, Serpent and the coupled reactor theory.
Figure 1. Serpent model of the Deuterium Critical Assembly

- (1) Heavy water external reflector.
- (2) Thermal core fuel lattice.
- (3) Heavy water moderator.
- (4) Internal heavy water reflector.
- (5) Air gap
- (6) Air above the thermal core
- (7) Steel (SUS340)
- (8) Heavy water.
- (9) Light water in the fast core
- (10) Fast core fuel lattice.
- (11) Aluminium tube in the middle of the fast core of wall thickness 0.2 mm. It contains air.

Figure 2: Criticality Factors and Coupling Coefficients at Different Light Water Levels in the Fast Core with the Heavy Water Level at 100 cm in the Thermal Core Calculated by Serpent, MCNP5 Codes and Coupled Reactor theory.

Criticality Coefficients and Coupling Coefficients

Criticality Coefficients

- $k_{eff}$ - Coupled Equation-MCNP
- $k_{eff}$ - Coupled Equation-Serpent
- $k_{eff}$ - MCNP
- $k_{eff}$ - Serpent

Coupling Coefficients

- $k_{11}$ - MCNP
- $k_{12}$ - MCNP
- $k_{11}$ - Serpent
- $k_{12}$ - Serpent
Figure 3: Multiplication Factors $k_{\text{eff}}$ at Different Light Water Levels with Heavy Water at 100 cm Calculated by Serpent and MCNP5 and the Coupled Reactor Theory.

Figure 4: Criticality Factors and Coupling Coefficients at Different Heavy Water Levels in the Thermal Core with the Light Water Level at 100 cm in the Fast Core, Calculated by Serpent, MCNP5 and the Coupled Reactor Theory.

Figure 5: Multiplication Factors $k_{\text{eff}}$, at Different Heavy Water Levels in the Thermal Core with the Light Water Level at 100 cm in the Fast Core, Calculated by Serpent and MCNP5 and the Coupled Reactor Theory.
Table 1: Average Percentage Difference between $k_{\text{eff}}$ as Calculated by MCNP5, Serpent and the Coupled Reactor Theory.

<table>
<thead>
<tr>
<th></th>
<th>%Δ$k_{\text{eff}}$ (MCNP5-Serpent)</th>
<th>%Δ$k_{\text{eff}}$ (MCNP5-Coupling theory)</th>
<th>%Δ$k_{\text{eff}}$ (Serpent-Coupling theory)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average percentage of difference $k_{\text{eff}}$ when setting the levels of heavy water at 100 cm and change the level of the light water from 0 to 200 cm</td>
<td>0.07%</td>
<td>0.65%</td>
<td>0.38%</td>
</tr>
<tr>
<td>Average percentage of difference $k_{\text{eff}}$ when setting the levels of light water at 100 cm and change the level of the heavy water from 0 to 200 cm</td>
<td>1.04%</td>
<td>0.54%</td>
<td>0.43%</td>
</tr>
</tbody>
</table>

4.2 GeNERG-DIME/TEC - University of Genova

4.2.1 Introduction and model presentation

The Genoa Nuclear Energy Research Group (GeNERG) is part of the Thermal Energetics and Environmental Conditioning Section (TEC) of the Mechanical Engineering, Energetics, Management and Transports Department (DIME) at the University of Genoa (UNIGE). Our research activities are focused on the innovative reactors study through neutronics and transmutation analysis in the frame of nuclear fuel cycle closure.

The present paper shows a summary of comparison done by GeNERG between two Monte Carlo based codes: SERPENT 1.1.19 and MCNP5/MONTEBURNS2.0.

Comparison has been performed on a configuration representing the Allegro MOX 75 MWt reactor: it was proposed by the European GoFastR project (Gas-Cooled Fast Reactor) in the frame of the 7th European Framework Program. Allegro is a gas cooled fast reactor with 75 MWt thermal power and high-pressure helium as coolant, it was simulated with standard fuel in the form of MOX24 pin S/A and Exp-S/A, these last to test GCFR fuel assemblies. SERPENT 1.1.19 was compared with MCNP5 for criticality analysis and with MONTEBURNS2.0 for burnup calculations. For comparison it was used the JEFF-3.1 cross sections dataset. Criticality parameters are set as follow:

- 5000 initial neutrons source;
- 500 active cycles;
- 25 inactive cycles;
- initial guess for $k_{\text{eff}}$.

Following figures show Allegro cross sections created by SERPENT 1.1.19.
Figure 6: Allegro X-Y cross section and particulars (left MOX24 pin S/A, right Exp-S/A) at active core height (created by SERPENT geometry-plotter).

Figure 7: Allegro Y-Z cross section (created by SERPENT geometry-plotter).
Figure 8: Allegro Y-Z cross section (created by SERPENT mesh-plotter)

Figure 9: Allegro X-Z cross section (created by SERPENT mesh-plotter)
4.2.2 Results

For sake of clearance, it is important to highlight that models for SERPENT 1.1.19 and MCNP5/MONTEBURNS2.0 are substantially identical as far as geometric, material and criticality parameters.

4.2.2.1 Static calculations

Static calculation times were comparable although longer for MCNP5 than SERPENT 1.1.19; however different statistical errors (standard deviations) were obtained. The criticality comparison is done on $k_{\text{eff}}$ (implicit) values:

- SERPENT: $k_{\text{eff}} = 1.04280$ with s.d. 0.00049
- MCNP: $k_{\text{eff}} = 1.04309$ with s.d. 0.00032

4.2.2.2 Burnup calculations

Burnup calculation times were higher for MONTEBURNS2.0 than SERPENT 1.1.19, however (again) different statistical errors (standard deviations) were obtained, as far as $k_{\text{eff}}$ values are concerned. In next Figure 11 it is shown the $k_{\text{eff}}$ vs. time trends; standard deviation is shown too.

Figure 10: Allegro X-Y cross section (created by SERPENT mesh-plotter)
Figure 11: Trends of $k_{eff}$ vs. time for burnup calculation.

In next Figure 12 it is shown the comparison between 6-group spectra for fuel pin loaded in MOX24 pin S/A, both in Begin of Cycle (BOC) and in End Of Cycle (EOC), i.e. after 967.25 burnup days. It is to note a very good agreement between values.

Figure 12. 6-group flux spectrum comparison for fuel pin.

In next Figure 13, the trends of U235 amount vs. burnup are shown. The comparisons concern a new figure of merit: the ratio between mass and fission cross section vs. burnup. Masses are considered for both SERPENT 1.1.19 and MONTEBURNS2.0 at begin of each step (BOS). It is to note a good agreement as far as the trends are concerned.
4.2.2.3 Conclusions

Results show a good agreement between SERPENT 1.1.19 and MONTEBURNS2.0/MCNP5 codes, however some differences exist:

- run times are different both for criticality and burnup calculations;
- statistical errors present different values, so that precision in calculation is not the same.

From criticality results point of view, \( k_{eff} \) trend vs. burnup shows some differences, however the maximum relative difference value is lower of 0.6%. 6-group spectra comparison shown a very good agreement both in BOC and in EOC. Also trends of figure of merit considered are in agreement, although some differences between values exist. In concluding of this comparison paper (summary), it can said that SERPENT 1.1.19 and MONTEBURNS2.0/MCNP5 are in good agreement both in criticality as burnup calculations. Depending on facility simulated and results researched, calculation times and precision in statistical errors can be considered more or less important. However a lower calculation times is preferred, above all if statistical precision obtained is already adequate to simulated problem.

4.3 SECNRS

4.3.1 Introduction

This paper are shown results of Serpent 1.1.19 verification for criticality calculations on the basis of a benchmark containing analysis NPP systems of fresh fuel supply and spent fuel treatment services [17] are presented.

General tasks is to study difference between the low-enrichment uranium (LEU) and mixed-oxide (MOX) fuel criticality safety issues and calculate the possible problematic variants. The assembly is loaded with several different types of fuel pins with different fuel enrichments. The LEU assembly contains 3.7-wt % and 4.2-wt %-enriched (in 235U) fuel pins, as well as uranium-gadolinium fuel pins. The MOX assembly contains 2.4-wt %, 2.7-wt % and 3.6-wt %-
enriched (in 239Pu) fuel pins, as well as uranium-gadolinium fuel pins. The pin loading for LEU and MOX assemblies are shown in Fig. 14.

![LEU and MOX fuel assembly](image)

Figure 14: VVER-1000 LEU and MOX fuel assembly.

### 4.3.2 Description of experiments

First series of experiments (series A) describe fresh fuel storage inside the plant. Fresh fuel is received at the plant by rail with units stacked on railroad cars. Two stainless steel canisters are welded together to a support structure to form a so-called TUK (package-set). Each package unit contains one assembly (MOX or LEU). A previous study [18] has shown that a water density of 0.2 g/cm³ between the canisters in an array of canisters containing VVER assemblies yields the highest $k_{\text{eff}}$. This situation might correspond to some type of fire-suppression condition. To determine if a critical array could exist, an infinite-array calculations are performed.

Second series of experiments (series B) describe fresh fuel transport within plant. Before the fuel is loaded to the reactor, the TUKs are up-ended and opened. Fresh fuel assemblies are then removed and placed in a transportation device called the fresh fuel transportation vehicle (FTV). The FTV has a height of 567 cm, an inside diameter of 200 cm, and a wall thickness of 30 cm. Assemblies are assumed to rest on the floor of the FTV. The FTV can hold 18 assemblies. However, only 16 of these 18 positions are filled with assemblies. In the center of the FTV, there is a hexagonal support structure that is also used for capturing. The center-to-center pitch measured along the flat-to-flat distance of the assemblies is 40 cm. Radial and axial cross-sectional views of the FTV containing full array of 16 LEU assemblies are shown in Figs. 15 and 16, respectively.
Fig. 15. Sixteen LEU assemblies in the fuel transportation vehicle.

Figure 16: Sixteen LEU assemblies in the fuel transportation vehicle – side view.
Calculated $k_{\text{eff}}$ for various cases of filling, such as:

- full array of 16 fresh LEU or MOX assemblies (Fig. 15);
- array of 13 fresh LEU assemblies, except the middle three are replaced by MOX assemblies (Fig. 17);
- full array of 18 fresh LEU or MOX assemblies (Fig. 18);
- array of 15 fresh LEU assemblies, except the middle three are replaced by MOX assemblies;

*Figure 17: Thirteen LEU and three MOX assemblies in the fuel transportation vehicle.*

*Figure 18: Eighteen LEU assemblies in the fuel transportation vehicle.*

Cases B1–B3 and B1m-B3m do not contain any water. Cases B4–B6 and B4m-B6m are the same configurations as B1–B3 and B1m-B3m, except the interstitial regions in the FTV and
the fuel assemblies are occupied with full-density water. For cases B7–B9 and B7m-B9m the water density is assumed to be 0.2 g/cm³. For cases B10 and B10m the water density is assumed to be 0.1 g/cm³. For cases B11 and B11m the water density is assumed to be 0.3 g/cm³. Cases B4–B11 and B4m-B11m are fully reflected by water with the same density as the water in the FTV.

Third series of experiments (series C) describe storage pool. The spent nuclear fuel storage pool holds 400 fuel assemblies having a total fuel weight of 165 MT. The pool measures 1325 cm long, 621 cm wide, and 1620 cm deep. The fuel assemblies are stored in hexagonal canning tubes made of 1% borated-stainless steel, which makes it possible to increase the capacity of the storage pool by a factor of 2 over the original pool design capacity. The calculated values for the design of the spacing of spent fuel in the spent fuel storage pool are based on a burnup of 40 MWd/kg for VVER-1000 spent fuel. The storage pool operates under atmospheric pressure.

The $k_{eff}$ for five cases are calculated, namely:

- $32 \times 17 \times 1$ array (total 544) of LEU assemblies in storage pool;
- $24 \times 16 \times 1 + 16 \times 1 \times 1$ array (total 400) of LEU assemblies in storage pool;
- $32 \times 12 \times 1 + 16 \times 1 \times 1$ array (total 400) of LEU assemblies in storage pool;
- $24 \times 16 \times 1 + 16 \times 1 \times 1$ array (total 400) of LEU assemblies with 3 MOX assemblies in the middle in storage pool (Fig. 19);
- Single LEU assembly in storage pool;

![Figure 19: Assemblies in the spent nuclear fuel storage pool in configuration 24×16 + 16×1.](image-url)

4.3.3 Description of model

The geometry and material composition of all configurations in the models were defined the same way as in the description of the benchmark [17].

Calculations for TUKs and FTV were performed with 750 active cycles and 30000 neutron sources per cycle. For storage pool because of the large size calculations were performed with 750 active cycles and 1 million neutron sources per cycle. Also for better accuracy, unresolved resonance probability tables were used for each calculation.
4.3.4 Results

Calculated $k_{\text{eff}}$ values are presented in Tables 2-5. The statistical uncertainty in the calculations are shown in tables.

Table 2. Calculated results for first series

<table>
<thead>
<tr>
<th>Description</th>
<th>$k_{\text{eff}}$ (SCALE)</th>
<th>$k_{\text{eff}}$ (Serpent)</th>
</tr>
</thead>
<tbody>
<tr>
<td>A1</td>
<td>LEU assemblies in TUKs optimum array search; infinite array; $\rho_{\text{water}} = 1.0 \text{ g/cm}^3$</td>
<td>$0,5141 \pm 0,0007$</td>
</tr>
<tr>
<td>A2</td>
<td>LEU assemblies in TUKs optimum array search; infinite array; $\rho_{\text{water}} = 0.2 \text{ g/cm}^3$</td>
<td>$0,7484 \pm 0,0007$</td>
</tr>
</tbody>
</table>

Table 3. Calculated results for second series (16 assemblies)

<table>
<thead>
<tr>
<th>Description</th>
<th>$k_{\text{eff}}$ (SCALE)</th>
<th>$k_{\text{eff}}$ (Serpent)</th>
</tr>
</thead>
<tbody>
<tr>
<td>B1</td>
<td>16 LEU assemblies in fuel transportation vehicle; dry</td>
<td>$0,2805 \pm 0,0002$</td>
</tr>
<tr>
<td>B2</td>
<td>16 MOX assemblies in fuel transportation vehicle; dry</td>
<td>$0,3061 \pm 0,0002$</td>
</tr>
<tr>
<td>B3</td>
<td>13 LEU and 3 MOX assemblies in fuel transportation vehicle; dry</td>
<td>$0,2867 \pm 0,0003$</td>
</tr>
<tr>
<td>B4</td>
<td>16 LEU assemblies in FTV; full-density water in interstitial regions and in the fuel assemblies</td>
<td>$0,8704 \pm 0,0007$</td>
</tr>
<tr>
<td>B5</td>
<td>16 MOX assemblies in FTV; full-density water in interstitial regions and in the fuel assemblies</td>
<td>$0,8440 \pm 0,0008$</td>
</tr>
<tr>
<td>B6</td>
<td>13 LEU and 3 MOX assemblies in FTV; full-density water in interstitial regions and in the fuel assemblies</td>
<td>$0,8675 \pm 0,0007$</td>
</tr>
<tr>
<td>B7</td>
<td>16 LEU assemblies in FTV; low-density (0.2 g/cm$^3$) water in interstitial regions and in the fuel assemblies</td>
<td>$0,9342 \pm 0,0002$</td>
</tr>
<tr>
<td>B8</td>
<td>16 MOX assemblies in FTV; low-density (0.2 g/cm$^3$) water in interstitial regions and in the fuel assemblies</td>
<td>$0,8731 \pm 0,0007$</td>
</tr>
<tr>
<td>B9</td>
<td>13 LEU and 3 MOX assemblies in FTV; low-density (0.2 g/cm$^3$) water in interstitial regions and in the fuel assemblies</td>
<td>$0,9189 \pm 0,0006$</td>
</tr>
<tr>
<td>B10</td>
<td>16 LEU assemblies in FTV; low-density (0.1-g/cm3) water in interstitial regions and in the fuel assemblies</td>
<td>$0,8903 \pm 0,0008$</td>
</tr>
<tr>
<td>B11</td>
<td>16 LEU assemblies in FTV; low-density (0.1-g/cm3) water in interstitial regions and in the fuel assemblies</td>
<td>$0,8912 \pm 0,0007$</td>
</tr>
</tbody>
</table>
Table 4. Calculated results for second series (18 assemblies)

<table>
<thead>
<tr>
<th>Description</th>
<th>( k_{\text{eff}} ) (SCALE)</th>
<th>( k_{\text{eff}} ) (Serpent)</th>
</tr>
</thead>
<tbody>
<tr>
<td>B1m</td>
<td>18 LEU assemblies in fuel transportation vehicle; dry</td>
<td>0.3001 ± 0.0003</td>
</tr>
<tr>
<td>B2m</td>
<td>18 MOX assemblies in fuel transportation vehicle; dry</td>
<td>0.3262 ± 0.0002</td>
</tr>
<tr>
<td>B3m</td>
<td>15 LEU and 3 MOX assemblies in fuel transportation vehicle; dry</td>
<td>0.3060 ± 0.0003</td>
</tr>
<tr>
<td>B4m</td>
<td>18 LEU assemblies in FTV; full-density water in interstitial regions and in the fuel assemblies</td>
<td>0.8724 ± 0.0007</td>
</tr>
<tr>
<td>B5m</td>
<td>18 MOX assemblies in FTV; full-density water in interstitial regions and in the fuel assemblies</td>
<td>0.8469 ± 0.0008</td>
</tr>
<tr>
<td>B6m</td>
<td>15 LEU and 3 MOX assemblies in FTV; full-density water in interstitial regions and in the fuel assemblies</td>
<td>0.8736 ± 0.0008</td>
</tr>
<tr>
<td>B7m</td>
<td>18 LEU assemblies in FTV; low-density (0.2 g/cm(^3)) water in interstitial regions and in the fuel assemblies</td>
<td>0.9826 ± 0.0007</td>
</tr>
<tr>
<td>B8m</td>
<td>18 MOX assemblies in FTV; low-density (0.2 g/cm(^3)) water in interstitial regions and in the fuel assemblies</td>
<td>0.9144 ± 0.0007</td>
</tr>
<tr>
<td>B9m</td>
<td>15 LEU and 3 MOX assemblies in FTV; low-density (0.2 g/cm(^3)) water in interstitial regions and in the fuel assemblies</td>
<td>0.9674 ± 0.0006</td>
</tr>
<tr>
<td>B10m</td>
<td>18 LEU assemblies in FTV; low-density (0.1 g/cm(^3)) water in interstitial regions and in the fuel assemblies</td>
<td>0.9263 ± 0.0007</td>
</tr>
<tr>
<td>B11m</td>
<td>18 LEU assemblies in FTV; low-density (0.1 g/cm(^3)) water in interstitial regions and in the fuel assemblies</td>
<td>0.9296 ± 0.0007</td>
</tr>
</tbody>
</table>

Table 5. Calculated results for third series.

<table>
<thead>
<tr>
<th>Description</th>
<th>( k_{\text{eff}} ) (SCALE)</th>
<th>( k_{\text{eff}} ) (Serpent)</th>
</tr>
</thead>
<tbody>
<tr>
<td>C1</td>
<td>32 × 17 × 1 array (total 544) of LEU assemblies in storage pool</td>
<td>0.7063 ± 0.0007</td>
</tr>
<tr>
<td>C2</td>
<td>24 × 16 × 1 + 16 × 1 × 1 array (total 400) of LEU assemblies in storage pool</td>
<td>0.7080 ± 0.0007</td>
</tr>
<tr>
<td>C3</td>
<td>32 × 12 × 1 + 16 × 1 × 1 array (total 400) of LEU assemblies in storage pool</td>
<td>0.7072 ± 0.0007</td>
</tr>
<tr>
<td>C4</td>
<td>24 × 16 × 1 + 16 × 1 × 1 array (total 400) of LEU assemblies with 3 MOX assemblies in the middle in storage pool</td>
<td>0.7090 ± 0.0008</td>
</tr>
<tr>
<td>C5</td>
<td>Single LEU assembly in storage pool</td>
<td>0.6954 ± 0.0008</td>
</tr>
</tbody>
</table>
4.3.5 Conclusion

Presented results show a good accuracy of Serpent in case systems of fresh fuel supply and spent fuel treatment services. The average deviation of \(k_{\text{eff}}\) values calculated by Serpent from calculated by SCALE ones is about 0.7% and the maximum deviation does not exceed 2.3%.

4.4 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.

During 2012, more calculations were performed at EDF with the SERPENT code for the new EPR© reactor. We presents here two 2D calculations in HZP (Hot Zero Power) and CZP (Cold zero Power) conditions for the first cycle. Calculations performed show good agreement against EDF TRIPOLI© calculations and against the EDF industrial scheme (CASSIOPEE) which will be used for startup tests predictions and safety evaluation of reloaded cores. Calculations have been performed with JEF2, JEF3 and ENDFB7 to confirm the interest of the latest cross sections evaluation libraries for the modelization of the heavy reflector of the EPR® reactor. We also have modelized the full core benchmark (J. Eduard Hoogenboom et al.) to acquired 3D experience with SERPENT. Some figures are reported below.

In order to simplify the representation of the baffle and more widely a shape within we can insert assemblies of a commercial PWR, EDF introduce the "gcross" shape. The shape is represented below.
The SERPENT code was installed on our new CASANOVA cluster (more than one hundred intel xeon CPU) and the intel compiler version 13.1 with no problem.

*Figure 20: EPR© - CZP (left) and HZP (right) flux distributions*
Figure 21: EPR© - Relative discrepancies between JEF3 (reference) and JEFF2 for the CZP state

<table>
<thead>
<tr>
<th>Library</th>
<th>Difference</th>
<th>k\text{eff}</th>
</tr>
</thead>
<tbody>
<tr>
<td>JEF3</td>
<td>0.865440</td>
<td>-</td>
</tr>
<tr>
<td>JEFF2</td>
<td>0.866221</td>
<td>78</td>
</tr>
<tr>
<td>ENDFB7</td>
<td>0.866094</td>
<td>65</td>
</tr>
</tbody>
</table>

Figure 22: EPR© - Absolute discrepancies between JEF3 (reference) and ENDFB7 for the CZP state.

Figure 23: EPR© - Relative discrepancies between JEF3 (reference) and ENDFB7 for the HZP state.
5. Summary, conclusions and future plans

During 2012, the focus in Serpent development was clearly shifted from the publicly distributed Serpent 1 to the beta version of Serpent 2, which was made available to registered users at the beginning of the year. The work on Serpent 1 is not completely frozen, but code updates are limited to bug fixes and minor modifications in existing capabilities. One of the goals set for the development of the new version was the capability to extend burnup calculations from fuel assembly to core level, without limitations due to excessive memory consumption or parallelization. This goal has been considered achieved. Serpent 2 is currently capable of performing the same tasks as Serpent 1, and the public distribution of the new version is planned for late 2014.
The work carried out for developing new features in Serpent 2 can be roughly divided in two main topics:

1) Advanced method for spatial homogenization

2) Coupled multi-physics applications

The methods used for spatial homogenization are planned to be completed in 2013-2014, and significant effort will be devoted to validating these capabilities for conventional LWR fuel cycle and transient simulator calculations, using state-of-the-art nodal diffusion codes. In order to become a practical tool for group constant generation, more work is needed for the automation of the calculation sequence. In practice this means, for example, implementing a built-in solver for homogeneous diffusion flux for the calculation of ADFs in geometries without reflective boundary conditions surrounding the homogenized region (reflectors, assembly colorsets), and the capability to perform branch (coefficient) calculations as an integral part of the assembly burnup cycle.

After the feasibility has been demonstrated in conventional 2D lattice calculation problems, homogenization methods will be extended to 3D geometries. There has already been considerable interest within the user community in using Serpent for this task.

Multi-physics is still a relatively new field of applications for Serpent, and the development of the related methods was started in 2012. The work is currently focused on internal coupling using the FINIX fuel behaviour module [19] and the COSY light-weight thermal hydraulics solver, as well as the development of a universal multi-physics interface for external code coupling. The coupling is based on the separation of the geometry model from the thermal hydraulic state-point information, which for the code user means that the problem can be set up without any modifications in the main input files. The work continues, and the first results are expected in 2014.

Work on a photon transport simulation mode was started in 2011, but had to be dropped due to more urgent topics. The work will be continued in late 2013. The main motivation for extending the transport simulation to photons is the capability to model gamma heating for multi-physics applications, but once implemented, the photon transport mode opens new interesting possibilities in radiation shielding and dose rate calculations as well.

Acknowledgements

Descriptions of recent work were provided by Mohamed Hussein (Royal Military College of Canada), Davide Chersola (University of Genova), Andrei Kirikin (SEC NRS) and Nordine Kerkar (EDF).

References


