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Serpent 2 as an inventory calculation code

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Abstract

Serpent is a Monte Carlo reactor physics and burnup calculation code developed at VTT Technical Research Centre of Finland. The code version 1 is available in the OECD/NEA Databank and is currently used in 72 universities and research organizations around the world. For those already familiar with the Serpent code, it is the natural tool to be used for inventory calculations in fuel cycle analyses. Others might, for example, be interested in using its progressive burnup calculation properties to provide results for comparisons with other codes.

This poster and the meeting paper "Intelligent Nuclide Selection Capability in the Reactor Physics and Inventory Calculation Code Serpent 2" introduce new code features that increase the usability of Serpent as an inventory calculator. Basically, the new features somewhat automatize the choice of inventory nuclides for which nuclide-wise data is printed in Serpent output, and also provide for importing of material compositions from Serpent to COSI6.

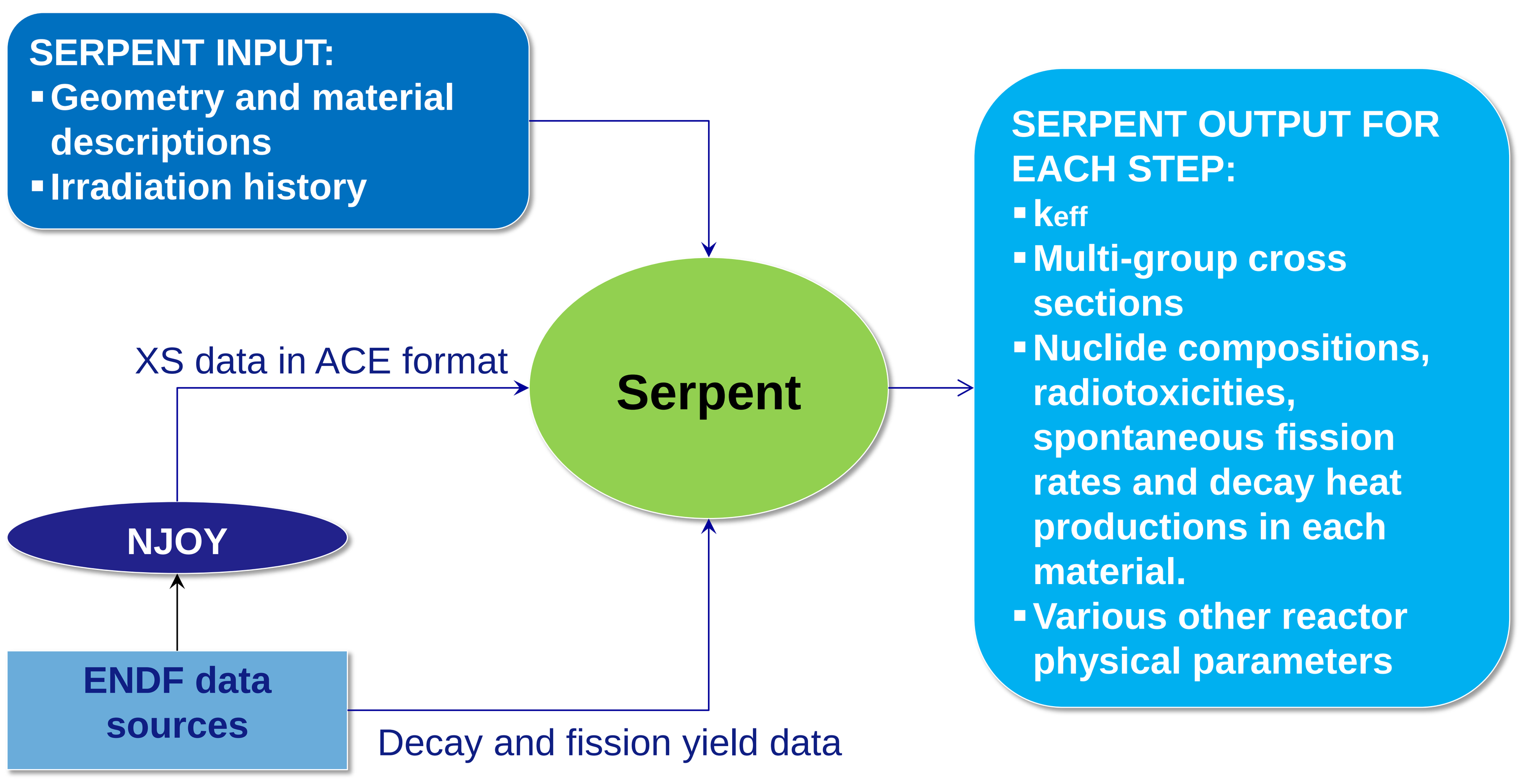
Serpent in a nutshell

Serpent is a reactor physics code with burnup calculation capability. Code version 1 is available in the OECD/NEA Data Bank while version 2, discussed in this paper, is only available to a limited group of beta testers.

Serpent uses ACE-formatted cross section data (same as in MCNP) that can be processed from the ENDF files using NJOY. Fission and decay data is read in ENDF file format. In burnup calculations, the Bateman equations are solved using the CRAM method. Serpent also utilizes predictor-corrector methodology in the time integration. The depletion output of Serpent includes atomic and mass densities, decay heat productions, inhalation and ingestion radiotoxicities together with spontaneous fission rates. The number of traceable nuclides depends on the library used, in the case of JEFF-3.1.1 about 1550 nuclides are tracked.

For more information, please visit Serpent website at

http://montecarlo.vtt.fi



New features for choosing inventory nuclides

Since printing out nuclide-wise results for all of the nuclides involved in the burnup calculation (~ 1500) would result in huge and clumsy output, nuclide-wise output is printed only for so-called inventory nuclides. Previously, these nuclides had to be entered manually in Serpent input, one by one. In this work, two new features are added to save manual work and to increase the usability of Serpent in inventory calculations:

- Series of nuclides can be added in the inventory by using hard-coded lists of important nuclides. The currently implemented lists include, for example, the nuclides used in the COSI6 fuel scenario code and nuclides of interest in long-term analyses.
- Top-*N* contributor nuclides of a quantity can now be automatically included in the inventory.

- \+ **Available in the OECD/NEA Databank**
- \+ **Full source code included**
- \+ **Burnup calculation with the CRAM method**
- \+ **Built-in Doppler pre-processor provides for accurate and easy modelling of temperatures**
- \+ **Fast, considering the method of solution**

- \- **No proton transport → spallation targets of ADSs must be modelled outside Serpent and included in the model as an external source**

Demonstration of the new top-*N* feature

The top-*N* feature was demonstrated by performing a burnup calculation for a PWR fuel assembly. It was burned to 60 MWd/kgHMi burnup using 44 burnup steps and then cooled for 100 000 years using 47 decay steps. The calculation time for the PWR case was 3 h 20 min when using 12 3.47 GHz Intel Xeon cores.

Results for top-5 decay heat producers in the PWR case are plotted in Figure 1.

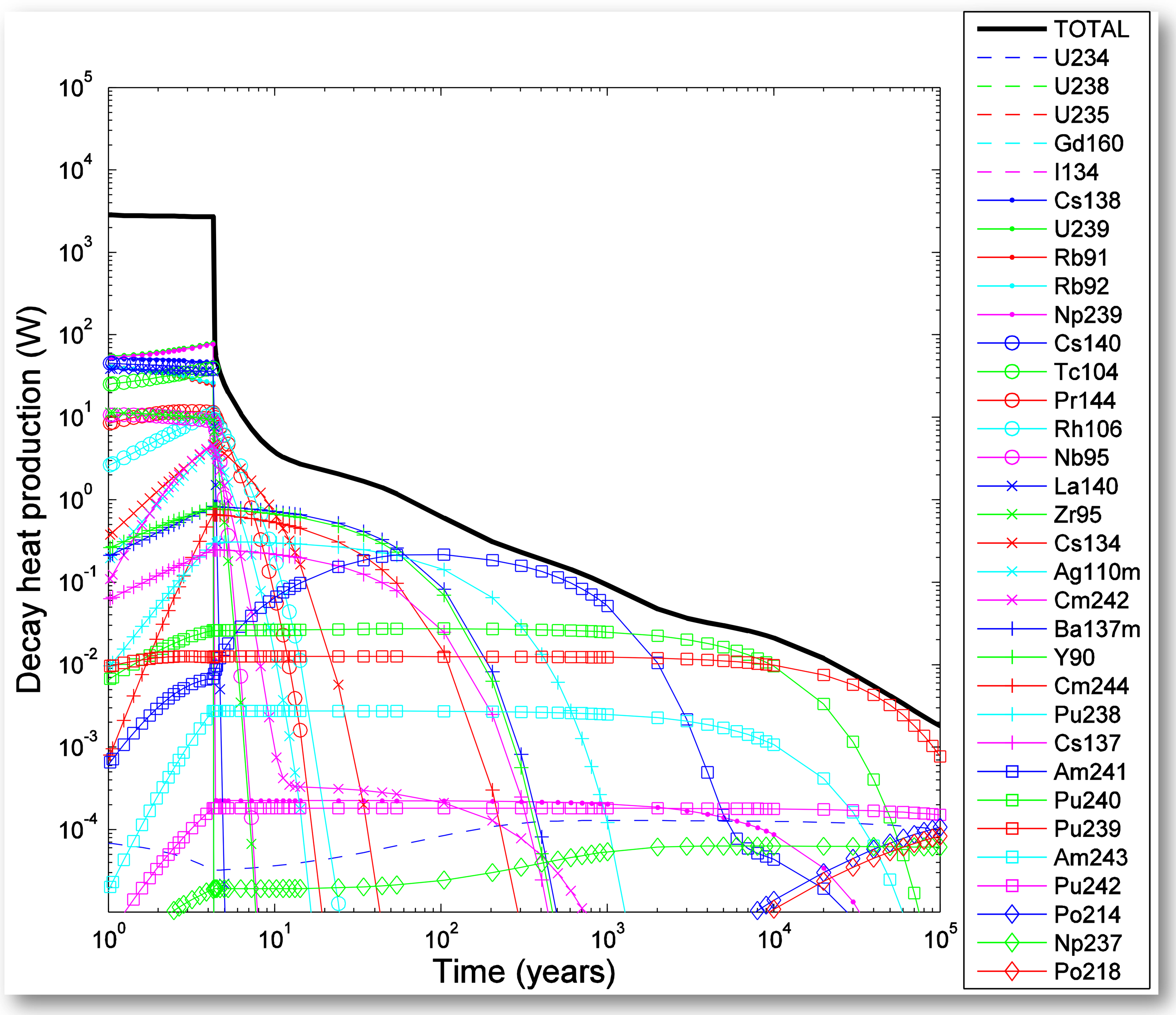


Figure 1: The total and nuclide-wise heat productions of the top-5 decay heat producers. The results are provided in Watts per axial centimetre of the assembly.