RACSA

Radiation Shielding and Criticality Safety Analyses

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Radiation shielding

Validating and testing the photon transport mode of Serpent

Reactor dosimetry

Validating Serpent for dosimetry applications, development of a new simplified dosimetry code

Criticality safety

Validation package for fresh fuel (Serpent, MCNP), improvement of burnup credit capabilities

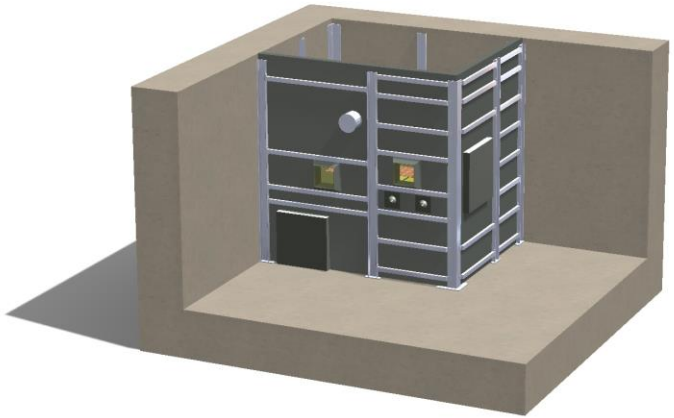
Radiation shielding

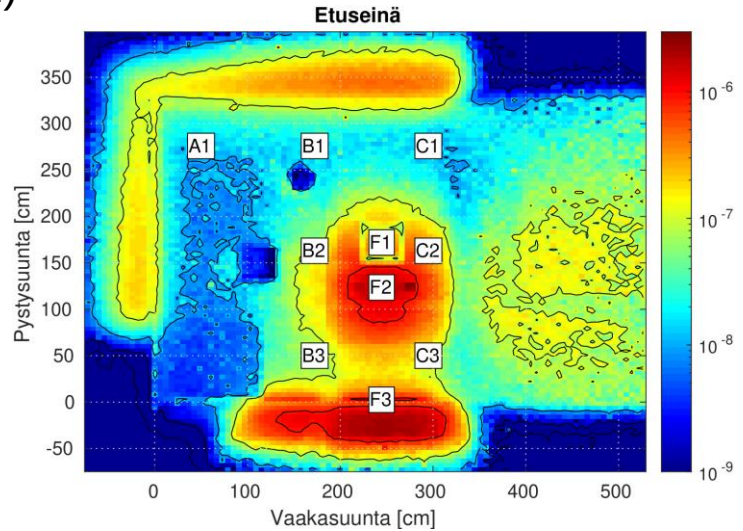
Starting point: photon transport and variance reduction methods implemented to Serpent previously, validation missing

- Validation cases from SINBAD and ICSBEP databases
  - OKTAVIAN: 14 MeV D-T neutron source, coupled neutron-photon transport, various shielding materials
  - ALARM: low-energy neutrons, photons with all energies ~<10 MeV
  - Good computational agreement with MCNP, varying agreement with experiments

Radiation shielding – hot-cell modelling

- VTT CNS hot-cell modelling with measurements
  - CAD-based geometry
  - Serpent photon transport in complex geometries, variance reduction required
  - Rather large discrepancies in C vs. E
  - (Not a code-to-experiment validation case!)





Reactor dosimetry

Required to evaluate the damage to the reactor pressure vessel caused by fast neutron fluence

- Previously: promising observations on Serpent's applicability for dosimetry analyses
  - Variance reduction functionality a key component
- Now: validation of Serpent for dosimetry applications
  - Pool Critical Assembly-Pressure Vessel Facility, H.B. Robinson-2 Pressure Vessel Dosimetry and NESDIP-2 Benchmark completed with satisfactory C/E results
  - Good Serpent vs. MCNP agreement in computational benchmarks
  - Further code development required
- New deterministic code under construction
  - Monte Carlo codes still not practical in multi-cycle neutron irradiation analyses
  - Preparatory work performed, will continue in 2021

Criticality safety

Validation package: required to define the systematic bias of the computing platform in criticality safety analysis for a known application (e.g. storage pool)

Previously achieved:

- Hundreds of critical experiments modelled for Serpent and MCNP to form the validation package
- Script to run the Serpent/MCNP calculations and perform the statistical analysis on the results

- Recently developed: tool to evaluate neutronic similarity between the application and the modelled experiments (for Serpent)
- Burnup credit – treat spent fuel as *spent* fuel
  - Effect of fission yield and decay data uncertainties on k_{eff} after irradiation, with Serpent → ~200 pcm (increases along with discharge burnup)

beyond
the obvious

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