# Recent spent fuel research at VTT

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# Outline

- Motivation and overview
- Results in selected topics
- Nuclear data uncertainty propagation
- Key take aways and future work

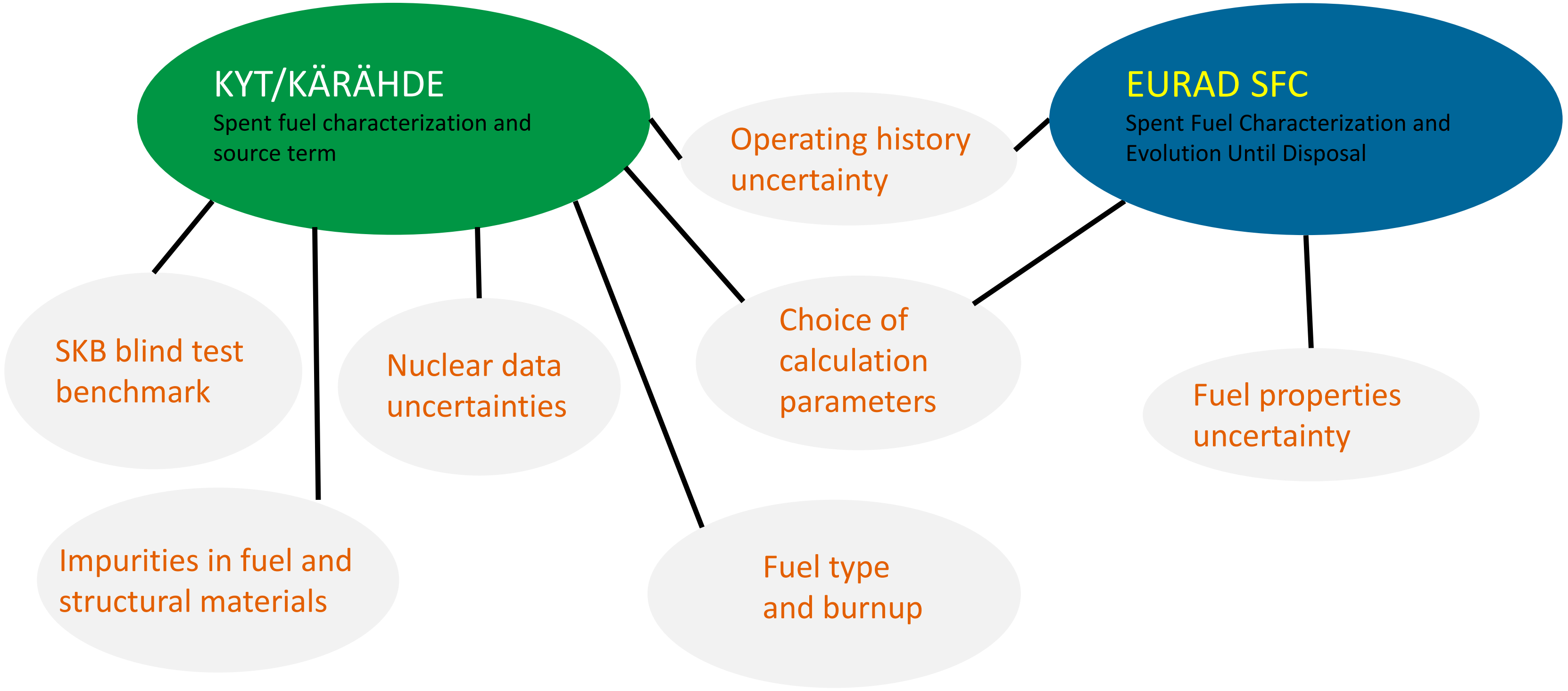
# Motivation and overview

# Motivation

- Knowledge of spent fuel (SNF) properties is important for safe handling and final disposal
  - Decay heat and reactivity → volume of repository space
  - Nuclide inventories → radiation protection, dose estimates
- Computational determination of decay heat involves several uncertainty sources
- SNF properties are also dependent on fuel type, enrichement and burnup
- At VTT, these uncertainties have been studied using the Monte Carlo reactor physics code Serpent 2 [1]

[1] Leppänen, J., et al. (2015) "The Serpent Monte Carlo code: Status, development and applications in 2013." Ann. Nucl. Energy, 82 (2015) 142-150.

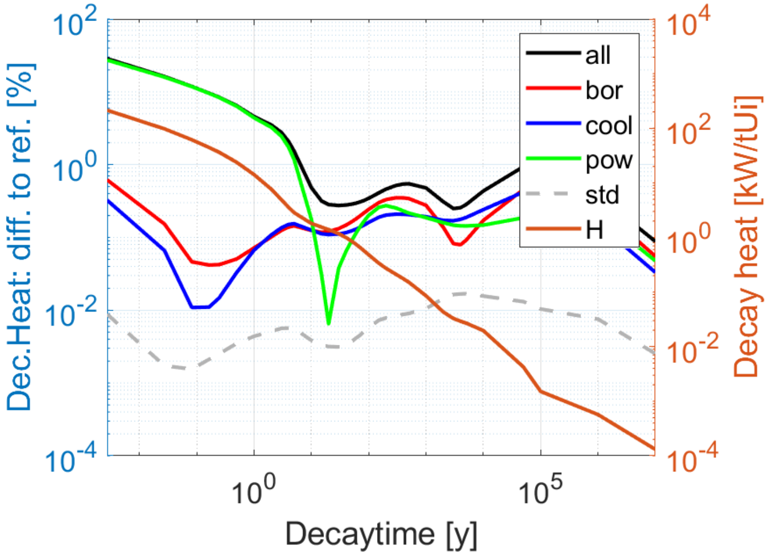
# Recent spent fuel studies at VTT

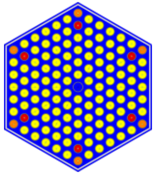


# Results in selected topics

# Simplifications in operating history

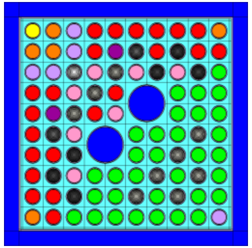
- Boron concentration, assembly power and coolant temperature and density were averaged over three cycles.
- Averaging assembly power overestimates activity, decay heat and photon emission rate 64 – 78 % right after irradiation.
- After 10 years the effect is < 1 %
- Effect on spontaneous fission rate and the studied nuclides (C-14, Cl-36, Mo-93, Ag-108m, I-129, U-233, U-235, Pu-239, Pu-241) is mostly below 1.5 %

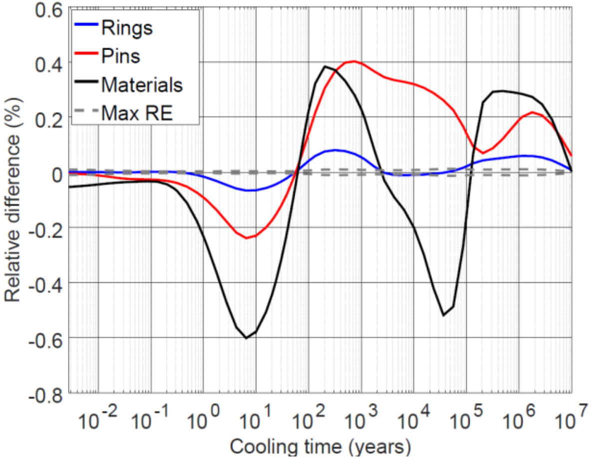


# Choice of calculation parameters

- Studied calculation parameters:
  - Depletion zone division
  - Depletion step length
  - Doppler-broadening rejection correction (DBRC)
  - Unresolved resonance probability table sampling
  - Energy-dependent branching ratios
- Maximum effects (depletion zone division)
  - Decay heat 0.6 % at 7 y
  - Photon emission rate 1.1 % at ~7000 y
  - Spontaneous fission rate 8 % at ~400 y
- Neglecting DBRC
  - non-negligible effect on some heavier nuclides
  - Maximum effect on DH and PE ~0.5 %

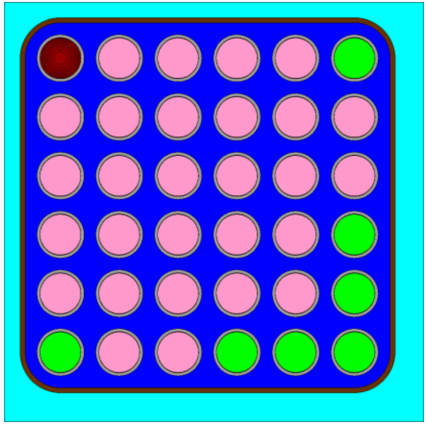





Decay heat

# Uncertainty analysis of one sample in a 6x6 BWR assembly

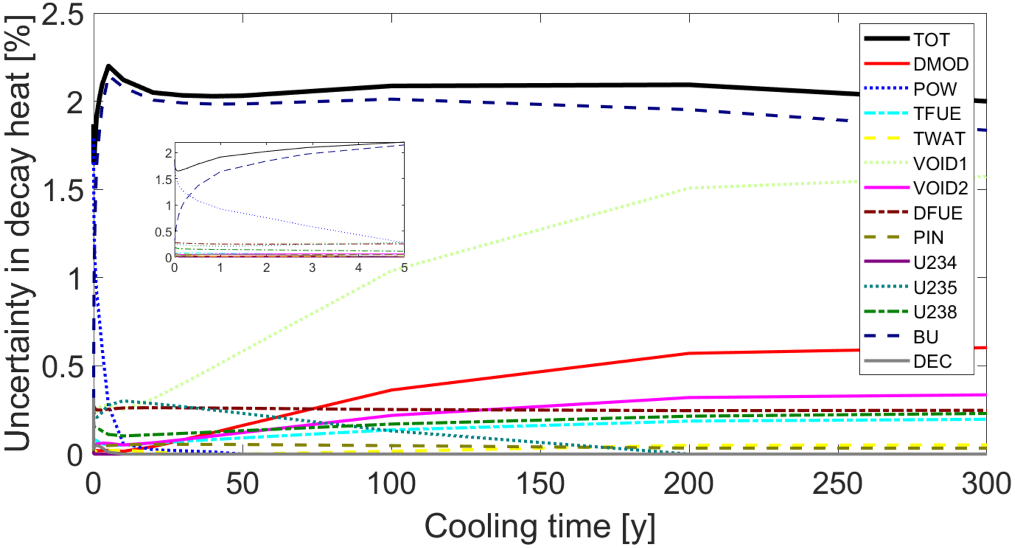
| Parameter | Average | RSD [%] |
|---|---|---|
| Moderator density | 0.759 g/cm$^3$ | 2 |
| Power density | 18.84 W/gU | 1.7 |
| Fuel temperature | 923 K | 2 |
| Water temperature | 549 K | 1 |
| Void$_1$ | 50 % | 6.7 (Δvoid = ±10 %) |
| Void$_2$ | 50 % | 1.3 (Δvoid = ±2 %) |
| Fuel density | 10.5 g/cm$^3$ | 0.3 |
| Fuel radius | 0.612 cm | 0.01 |
| $^{234}$U enrichment (in $^{235}$U) | 0.65 | 0.1 (Δ$^{234}$U = ±0.25 %) |
| $^{235}$U enrichment | 2.53 | 0.7 (Δ$^{235}$U = ±0.05 %) |
| $^{238}$U content | 85.92 | 0.7 |
| Burnup (assembly) | 22.627 | 2 |



$$\sigma_{calc,i}^2 = \sigma_i^2 + \sigma_{MC}^2$$

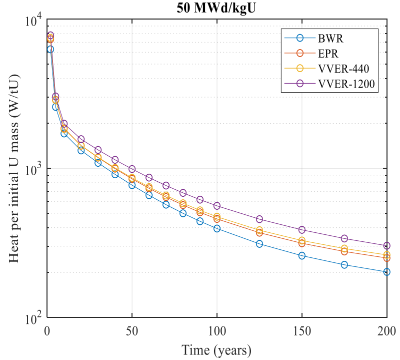
$$\sigma_{tot}^2 = \sum_{i=1}^{n} \sigma_i^2$$

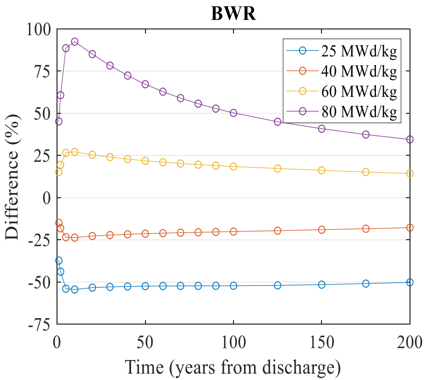
# Decay heat uncertainty of the BWR sample




| | | |
|---|---|---|
| TOT | – | Total uncertainty (void$_2$ = ±2 %) |
| DMOD | – | Moderator density |
| POW | – | Power density |
| TFUE | – | Fuel temperature |
| TWAT | – | Water temperature |
| VOID1 | – | Void$_1$ = ±10 % |
| VOID2 | – | Void$_2$ = ±2 % |
| DFUE | – | Fuel density |
| PIN | – | Fuel radius |
| U234 | – | $^{234}$U enrichment |
| U235 | – | $^{235}$U enrichment |
| U238 | – | $^{238}$U content |
| BU | – | Burnup |
| DEC | – | Decay data |

# Impact of fuel type and burnup on decay heat




- Difference to 50 MWd/kgU

Top decay heat contributors 5 y (top) and 100 y (below) after irradiation

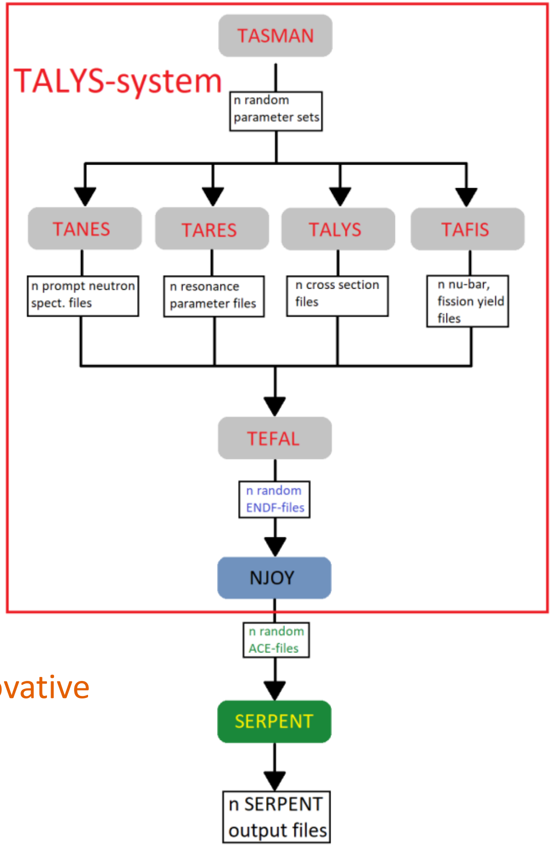
| BWR | % | EPR | % | VVER-440 | % | VVER-1200 | % |
|---|---|---|---|---|---|---|---|
| Y90 | 27.7 | Cs134 | 27.2 | Cs134 | 27.9 | Cs134 | 25.4 |
| Ba137m | 27.2 | Ba137m | 24.5 | Y90 | 25.2 | Y90 | 24.2 |
| Cs134 | 24.9 | Y90 | 23.2 | Ba137m | 25.1 | Ba137m | 23.2 |
| Cm244 | 10.2 | Cm244 | 13.7 | Rh106 | 11.4 | **Pu238** | **16.9** |
| Rh106 | 10.0 | Rh106 | 11.4 | Cm244 | 10.3 | Rh106 | 10.4 |

| Nuclides | % (BWR) | % (EPR) | % (VVER-440) | % (VVER-1200) |
|---|---|---|---|---|
| Am241 | 40.0 | 44.8 | 46.4 | 39.1 |
| Pu238 | 22.8 | 23.6 | 22.6 | **34.8** |
| Ba137m | 16.3 | 14.1 | 13.6 | 11.4 |
| Y90 | 15.0 | 12.1 | 12.4 | 10.7 |
| Pu240 | 5.9 | 5.4 | 5.0 | 4.0 |

# Nuclear data uncertainty propagation

# Key Idea

- Create hundreds of randomized nuclear data files for many nuclides
  - New nuclear data is produced from nuclear models and experimental data: No need to resort to covariance data.
  - The data is produced with TALYS and a few satellite programs = T6 [2]
- Assign a random combination of the random data files for a Serpent run
  - Repeat hundreds of times and compile uncertainty in the output with an appropriate statistic



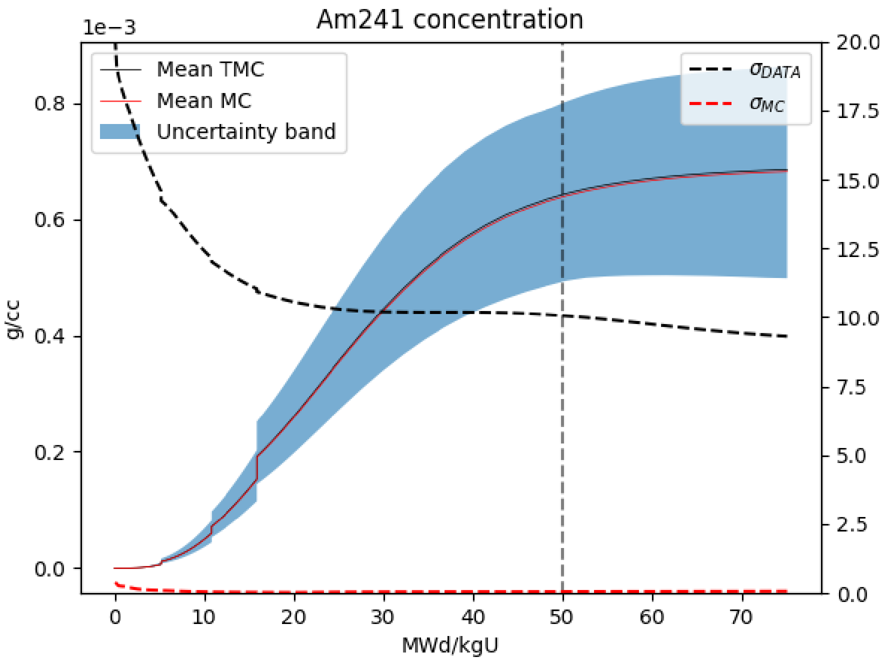

[2] A.J. Koning et al. (2019) “TENDL: Complete Nuclear Data Library for innovative Nuclear Science and Technology”. In: Nuclear Data Sheets 155, 1-55.

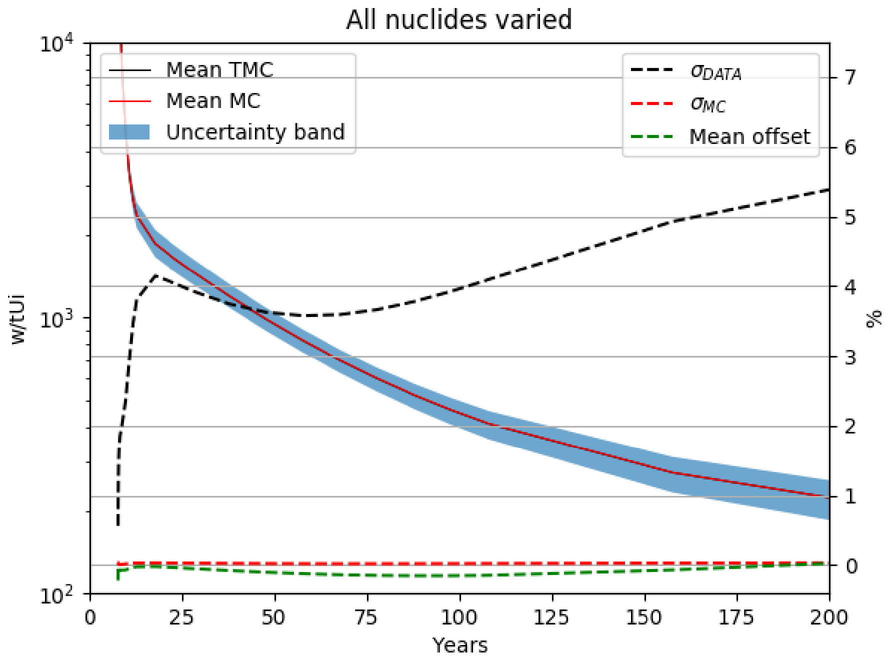
# Current state

- 500 files for 88 nuclides have been produced for 0 K, 300 K, 600K, 900 K and 1200 K temperatures
- Random fission yield files have been included
  - Adopted from TENDL – website
- Scripts for file processing, TALYS/Serpent environment initialization, data visualization etc. have been set up.
- It is now straightforward to propagate data uncertainties through Serpent, although computationally expensive.

# Significant Uncertainties in Actinide Concentrations

Uncertainty in decay heat > 3.5 % due to nuclear data when cooling time >15 y



# Key take aways and future work

# Key take aways

- Approximations in power density should be avoided, when dealing with ”fresh” SNF (cooling time < 10 y).
- Pin-wise depletion zone division should be used in burnup calculations and Gd pins should be further divided in radial rings.
- DBRC should not be neglected. Use on U isotopes and $^{239}$Pu and $^{240}$Pu is mainly sufficient.
- Significant sources of uncertainty for BWR assembly calculations
  - Nuclear data
  - Burnup
  - (Void fraction and moderator density)
- Using recycled uranium significantly increases decay heat from SNF

# Future work

- Full libraries and random nuclear data files with T6
- Expanding from 2D assembly calculations to 3D full core
  - Serpent – Ants sequence (Kraken computational framework [3])
- Participation in activities of NEA subgroup on decay heat

[3] Leppänen, J., et al. (2022) "Current Status and On-Going Development of VTT's Kraken Core Physics Computational Framework" Energies, 15, 876, https://doi.org/10.3390/en15030876.

# Selected publications

- International Journal publications
  1. Tuominen R., Valtavirta V. (2022), "The Effect of Serpent 2 Calculation Parameters on Evaluated Spent Nuclear Fuel Source Term", Journal of Nuclear Engineering and Radiation Science, 8(4), [044503], https://doi.org/10.1115/1.4051445. .
  2. Häkkinen S. (2022), "Impact of Approximations in Operating History Data on Spent Fuel Properties with Serpent 2", Journal of Nuclear Engineering and Radiation Science, 8(4), 041901. [NERS-21-1015]. https://doi.org/10.1115/1.4051444.
  3. Jansson P., et.al. (2022), " Blind Benchmark Exercise for Spent Nuclear Fuel Decay Heat", Nuclear Science and Engineering, 196, pp.1125-1145, https://doi.org/10.1080/00295639.2022.2053489.
  4. Rochman D., et.al., "On the estimation of nuclide inventory and decay heat: a review from the EURAD Eurpean project", Frontiers in Energy Research. (submitted in 2022).

- Conference articles
  1. Rintala A., "Evaluating the Effect of Decay and Fission Yield Data Uncertainty on Spent Nuclear Fuel Source Term Using Serpent 2", In Proc. 28th International Conference Nuclear Energy for New Europe, Paper No. 602, Portorož, Slovenia, 2019.
  2. Juutilainen P., Häkkinen S., "Impact of Fuel Type and Discharge Burnup on Source Term", In Proc. 28th International Conference Nuclear Energy for New Europe, Paper No. 906, Portorož, Slovenia, 2019.
  3. Rintala A., " Evaluating the Effect of Decay and Fission Yield Data Uncertainty on BWR Spent Nuclear Fuel Source Term", In Proc. 29th International Conference Nuclear Energy for New Europe, Paper No. 1506, Portorož, Slovenia, 2020.
  4. Tuominen R., Valtavirta V., "The Effect of Serpent 2 Calculation Parameters on Evaluated Spent Nuclear Fuel Source Term", In Proc. 29th International Conference Nuclear Energy for New Europe, Paper No. 1503, Portorož, Slovenia, 2020.
  5. Häkkinen S., "Impact of Approximations in Operating History Data on Spent Fuel Properties with Serpent 2", In Proc. 29th International Conference Nuclear Energy for New Europe, Paper No. 1505, Portorož, Slovenia, 2020.

- Research reports
  1. Rintala A., "Evaluating the effect of decay and fission yield data uncertainty on spent nuclear fuel source term using Serpent 2 – continued study", VTT Research Report, VTT-R-00209-20, 2020.
  2. Rintala A., "Separate effect of decay and fission yield data uncertainty on spent nuclear fuel source term", VTT Research Report, VTT-R-00106-21, 2021.
  3. Juutilainen P., "Effect of burnable absorber rods and U-235 enrichment on EPR UO2 fuel assembly source term with Serpent 2", VTT-R-00242-21, 2021.
  4. Häkkinen S., "Gundremmingen-A assembly B23 sample I2680 depletion calculation with Serpent 2", VTT Research Report, VTT-R-00631-21, 2021.
  5. Häkkinen S., "Sensitivity and uncertainty analysis of Gundremmingen-A assembly B23 sample I2680 depletion calculation with Serpent 2", VTT Research Report, VTT-R-00632-21, 2021.

- Thesis
  1. Vaara L., "Nuclear Data Uncertainty Propagation in Total Monte-Carlo Method", M.Sc. Thesis, Aalto University, 2022.
  2. Vaara L., "Impurity-generated impact on the characteristics of spent nuclear fuel", Special Assignment, Aalto University, 2020.

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# Acknowledgements


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# beyond the obvious

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