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Dose rate calculation at transport and storage casks for spent nuclear fuel

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  - Computational steps
  - Deterministic methods
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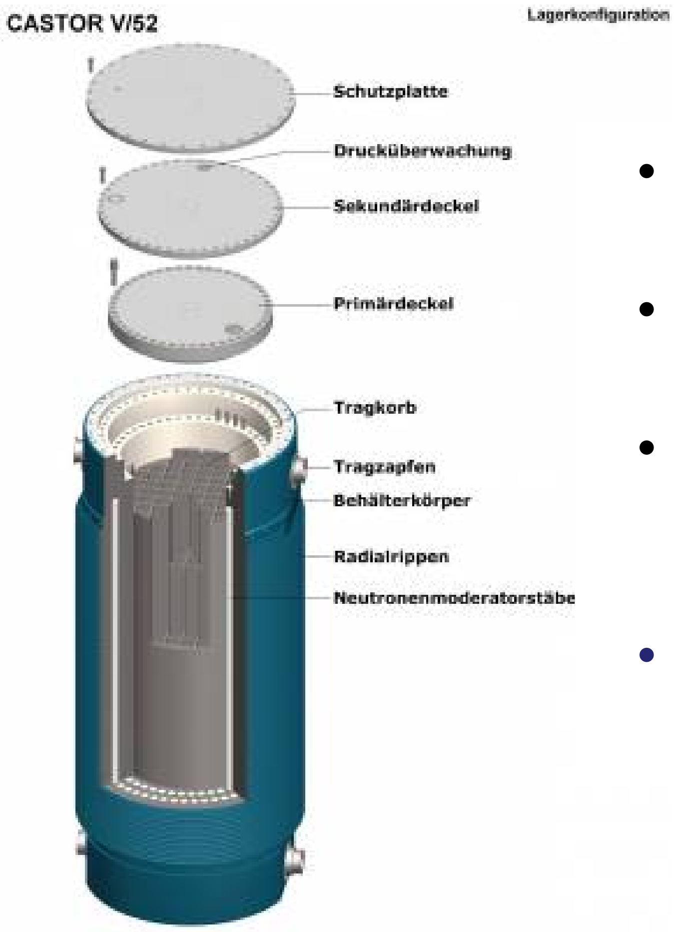
- Example of application: Dose rate at axial burn-up profile of fuel assemblies

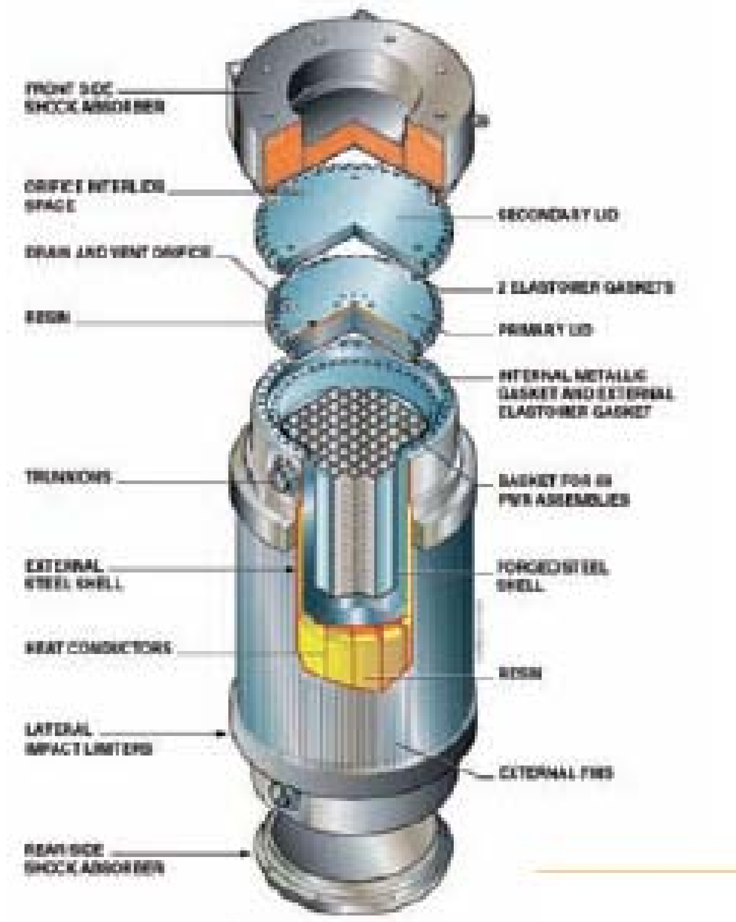
Interim storage of transport and storage casks

- Part of the German waste management agreement between government and nuclear industry
- 12 interim storage facilities at NPP sites and 3 central interim storage facilities are available
- TSCs capable for up to19 PWR or 52 BWR assemblies, corresponding to ~10 tons of heavy metal
- Assembly burn-ups up to 55 GWd/tHM and initial enrichments up to 4.6 wt% ^{235}U
- UO_2 and MOX fuel

TSC Main Safety Criteria for Design:



- **Maintaining Subcriticality**
- **Removal of Decay Heat**
- **Safe Enclosure of the radioactive Inventory**
- **Proper Shielding of Radiation**



TN 24 BH transport Configuration

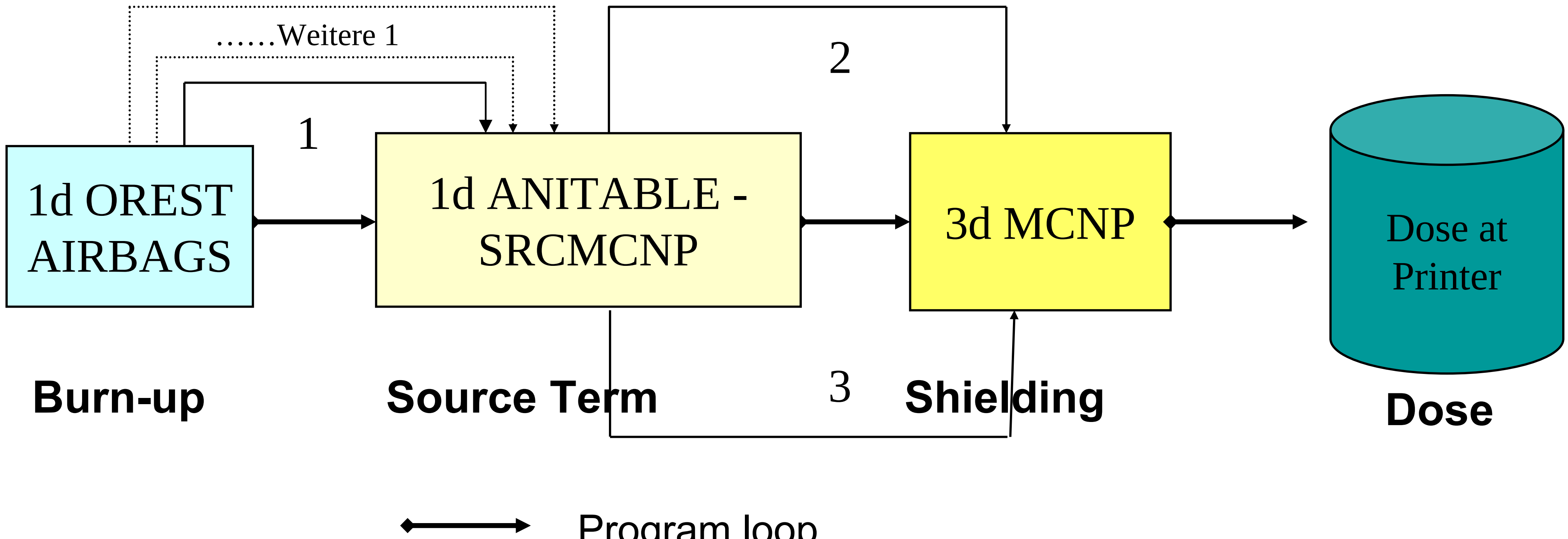
Calculation of the TSC dose rate: Four calculation steps

- Modeling of **burn-up** and **decay** of the radioactive inventories (approximately 1000 isotopes)
- Calculation of the **neutron** and **gamma spectrum**
- **Radiation transport** calculation for neutron and gamma radiation, starting inside the fuel through the TSC shell
- Calculation of **dose rate** at the cask surface and at selected distant points in the environment

GRS experience with systems for dose rate calculation of TSCs with spent fuel assemblies

- Since 1981 GRS developed systems for burn-up and dose calculation
- Application of 2- and 3-dimensional transport methods (2d DORT, 3d TORT) **and** statistical codes 3d MCNP
- Code systems support the user at the manyfold calculation steps
- Validated basis moduls are coupled by input / output routines
- So safety analyses of TSCs and interim storage arrays are possible

Coupling diagram of statistical 3d MCNP supported by GRS-ANITABLE - SRCMCNP



1. Axial inventories
2. n-γ Sources and axial burn-up profile
3. Dose conversion factors

Advantages and disadvantages of statistical methods for dose rate calculation

(+) 3d problems can be treated easier

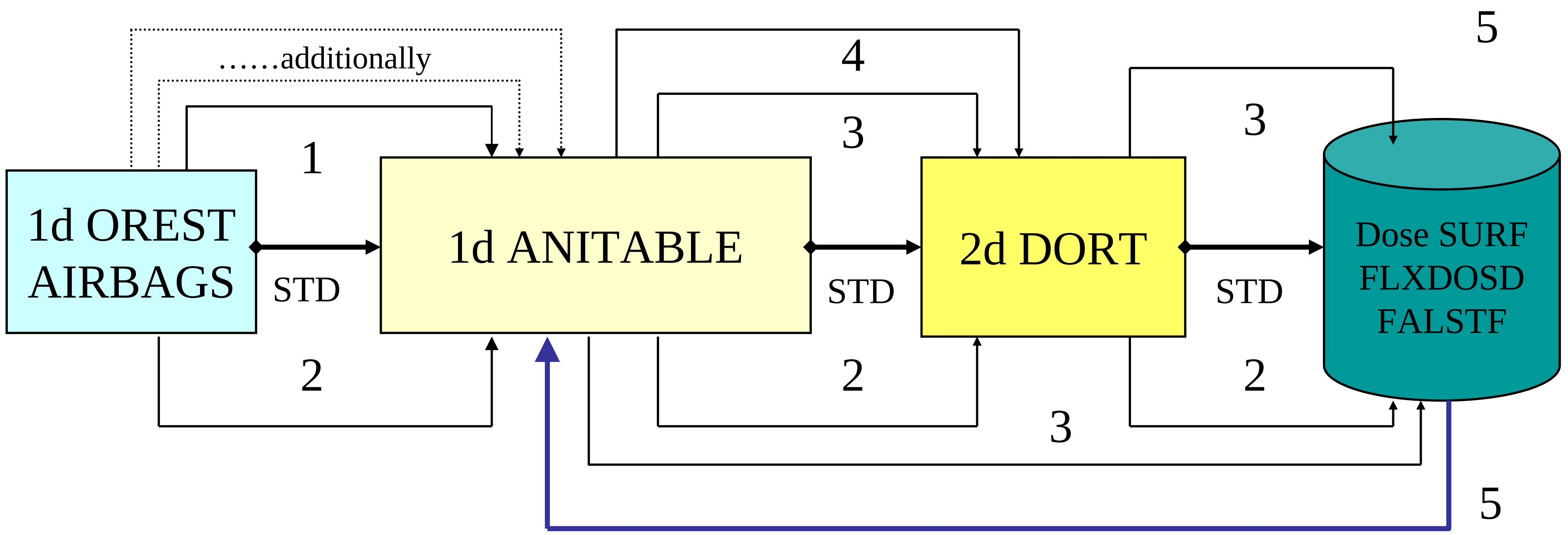
(+) Radiation transport in air no problem

(-) Flux and dose solution only at certain selected positions possible; so limited capabilities for further analysis

(-) relatively long computation times due to statistics

(-) modeling of extended source geometry can cause statistical problems

Coupling diagram of **deterministic** 2d DORTABLE supported by ANITABLE



➔STD: Program loop and transfer of basic data and results

1. Axial Inventories
2. Cross sections and fission source
3. Fluxes and dose conversion factors
4. n-γ-Sources with axial profile
5. ➔Feed back of mesh generator

Advantages and disadvantages of deterministic methods for dose rate calculation

(+) equally converged solution for flux in all space cells in „deep penetration“ problems, so a variety of further analysis is possible

(+) relatively short calculation time

(+) extended source areas can be treated well

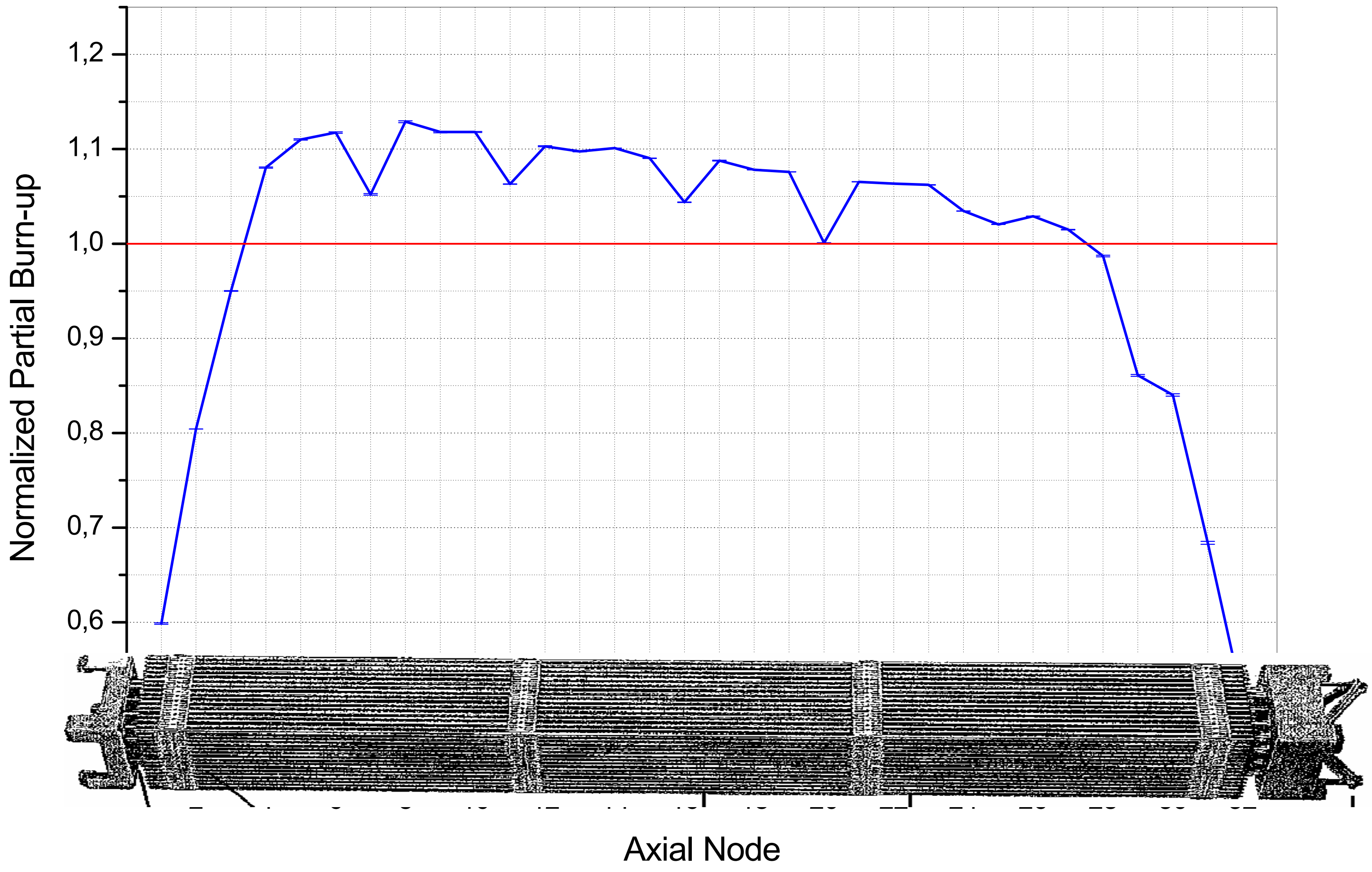
(-) Radiation transport in air shows „Ray“-Effect due to discrete meshes and solid angles

(-) 3d problems und 3d effects difficult to handle

Example of dose calculation of a generic TSC type CASTOR V/19® containing spent fuel elements

- 19 PWR assemblies type ‚Konvoi', initial enrichment 3,6 %
- Average burn-up 40 GWd/tSM, axial burn-up profile from 32 measured nodes, 10 years decay time
- Deterministic calculation by 2d code DORT (R-Z geometry)
- Comparison with results of 3d MCNP calculation using identical inventories and source terms

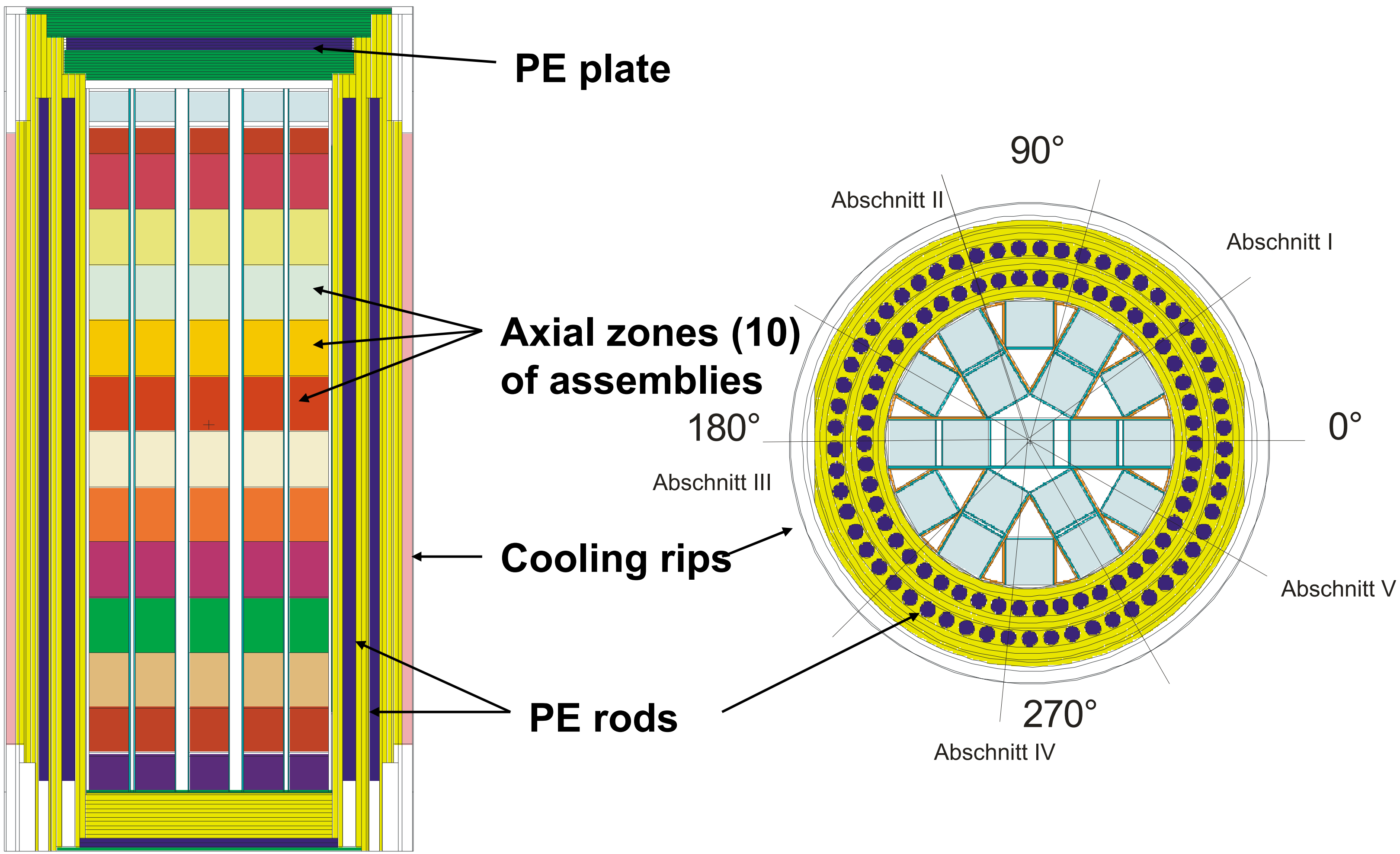
Axiale burn-up profile of a PWR fuel assembly in 32 nodes



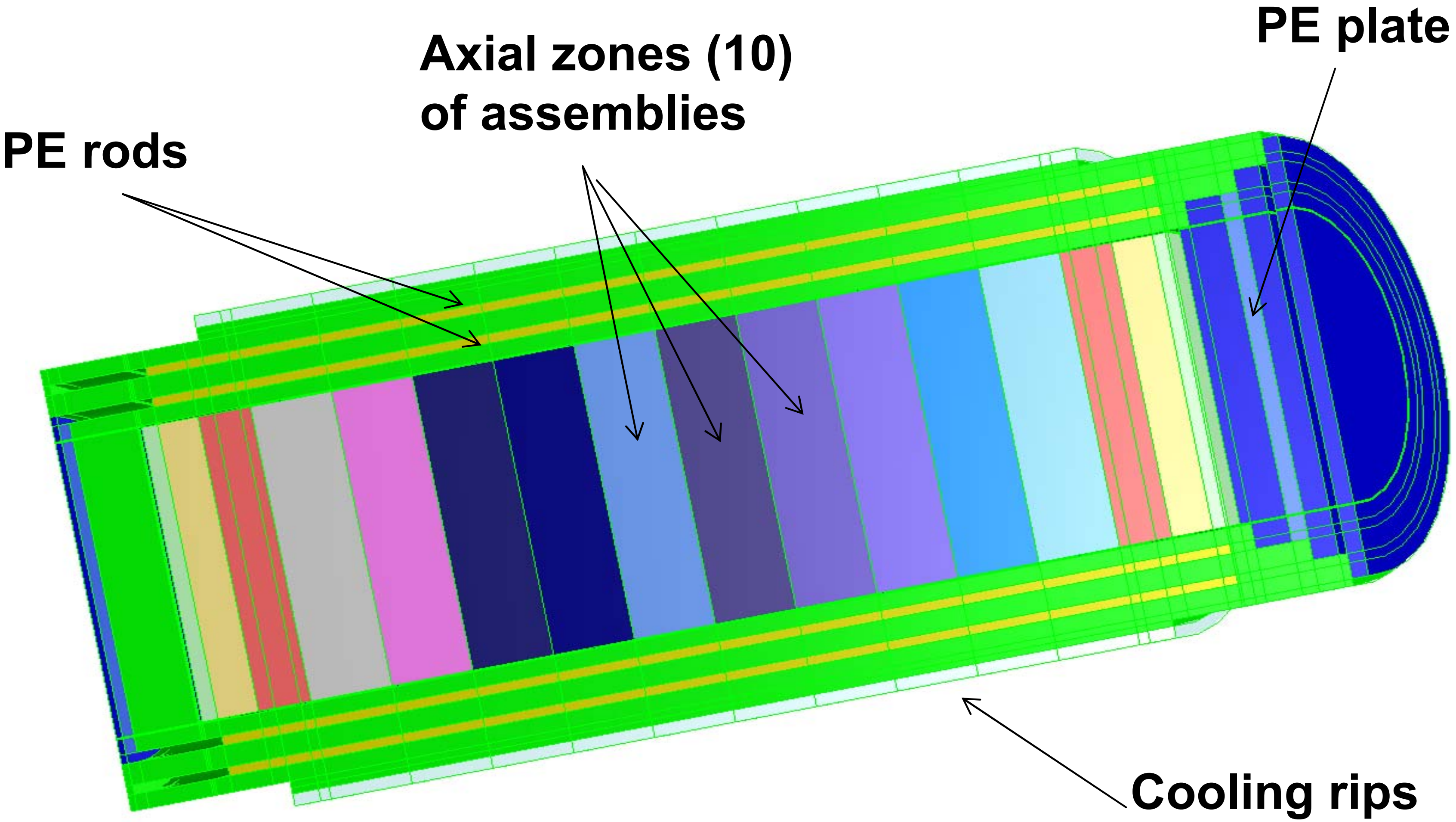
Inventory and Source term calculation for axial zones

- Approximation of the measured burn-up profile (32 nodes) by homogenized 10 axial zones to reduce calculation efforts
- Inventory calculation by OREST for 10-fold variation of the standard burn-up data with axial coolant distributions and fuel power distributions
- Transfer of axial inventories of 1000 isotopes and axial source terms to MCNP and DORTABLE:
- Source term distributions handled by weighting factors for the axial zones in the radiation transport codes
- Inventories collapsed to 50 main isotopes for axial cross sections

Geometric model of cask CASTOR V/19 for 3d MCNP



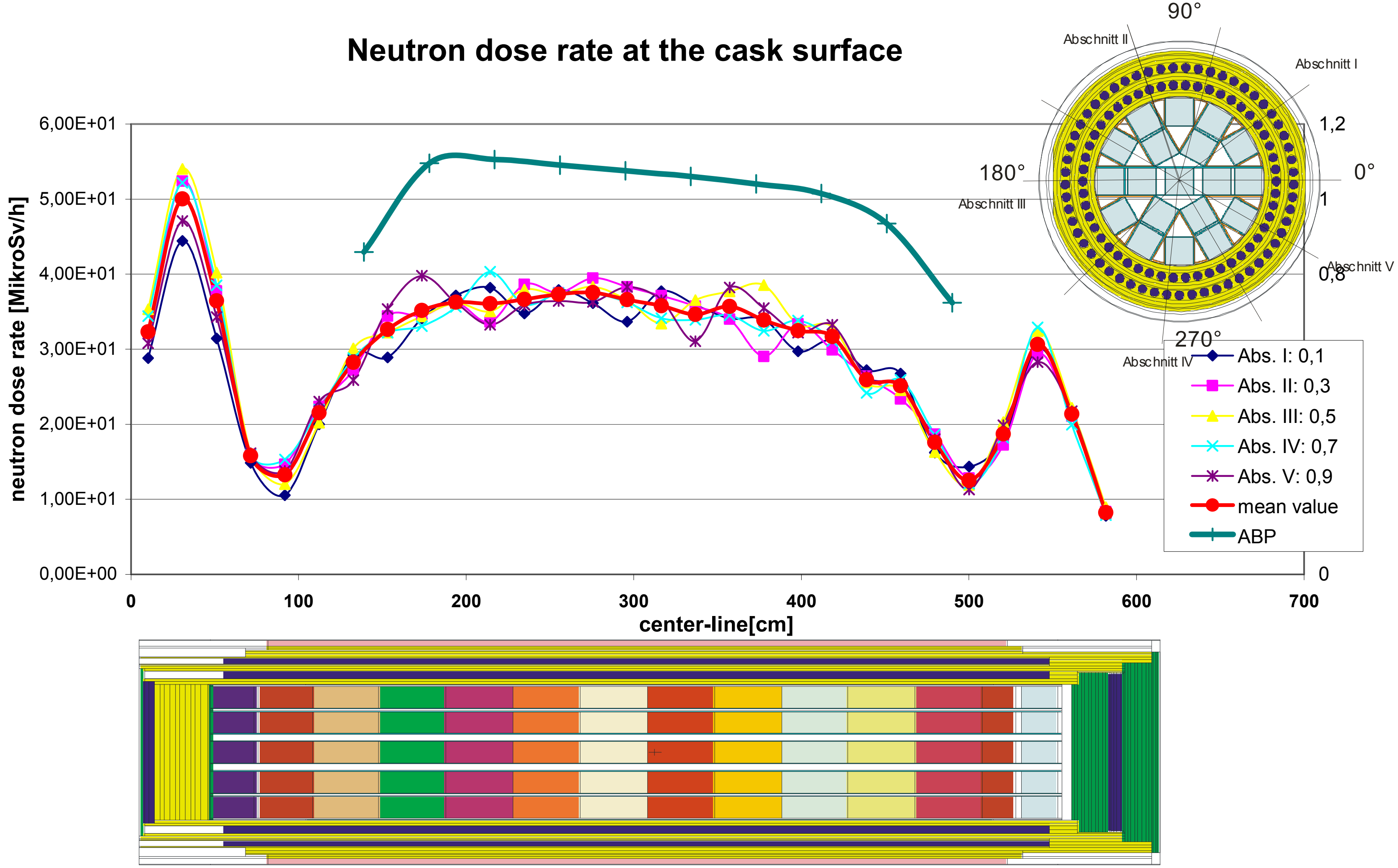
Geometric model of cask CASTOR V/19 for 2d DORTABLE



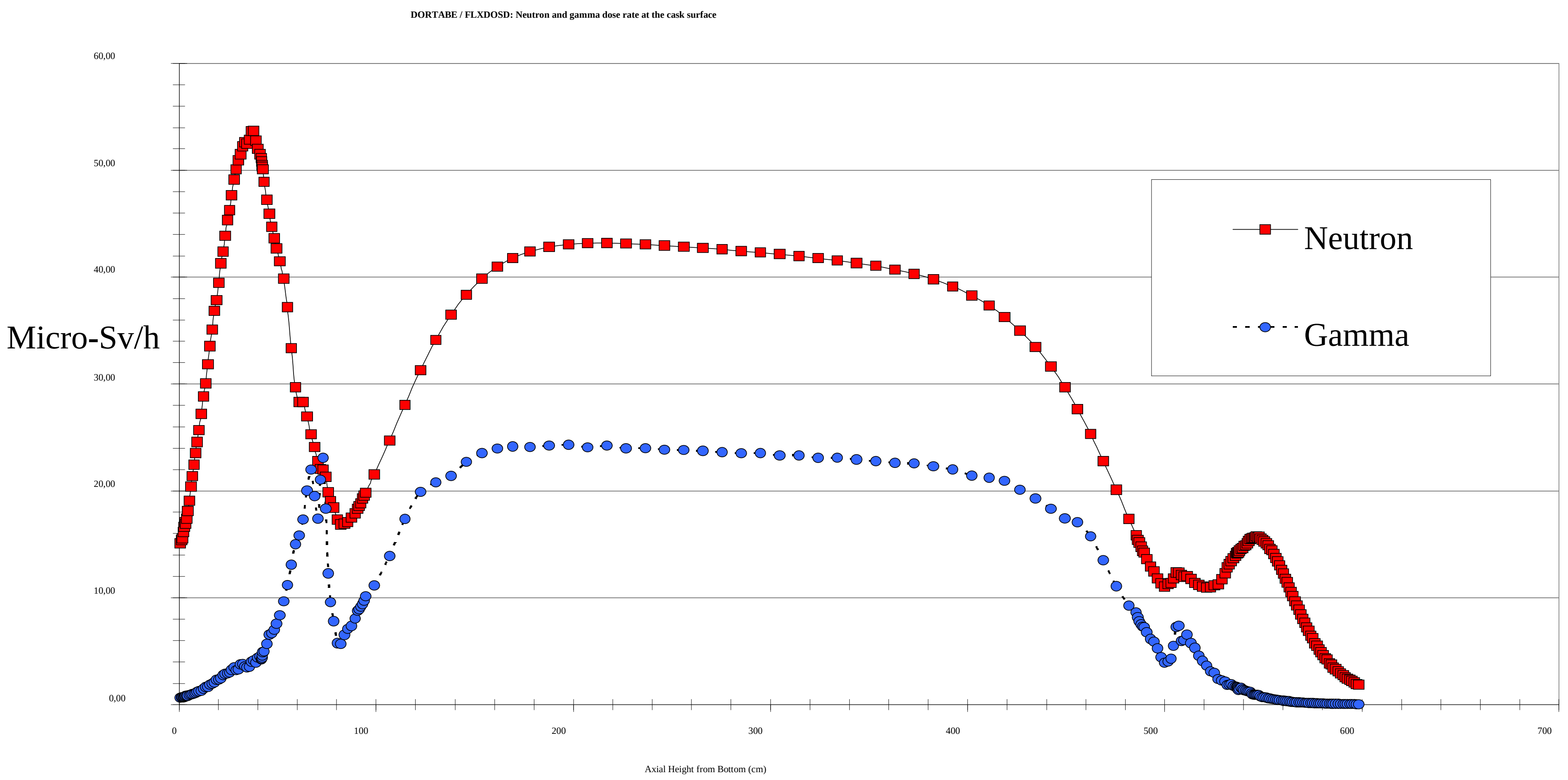
Special tasks and problems in TSC calculations

1. Axial distributions of TSC nuclide inventories concern simultaneously axially varying **cross sections and** axial **source** terms .This is important both for shielding and BUC problems
2. **Subcritical neutron multiplication** enhances both neutron dose and secondary n-γ-gamma dose by approx. 30 %
3. **PE-rods** in the cask shell have to be modeled as rings for 2d systems, solved by effectively calculated PE densities
4. **Proper mesh grid** calculation for SN codes ANISN, DORT and TORT in complex geometrical structures. Several start-up flux calculations are needed

Neutron dose results MCNP for CASTOR V/19 (µSv/h)



Neutron and gamma dose results DORT for CASTOR V/19 (µSv/h)



Summary and Outlook

- A physically satisfying, consistent method was found for the multifold problem of burn-up / source term / shielding / dose rate calculation by serially coupling of validated basic modules by user-supporting code systems
- A deterministic GRS system for 2d dose rate calculation at transport and storage casks in RZ geometry is available for application
- Deterministic and statistic methods show good agreement

Summary and Outlook (2)

- Effects of burn-up heterogenities can be modelled better for shielding and BUC tasks
- Considering burn-up profiles is not yet state of the art in shielding calculations of TSCs but maybe in near future
- Actually development to integrate DORTABLE in a larger system 'GRS-DORTACTIVE' for coupled fluency / activation / dose rate calculations of large reactor components during lifetime of a NPP
- Further development is the implementation of 3d TORT in a code system 'TORTABLE' is planned

Thank you for your attention