A Hybrid Deterministic / Stochastic Calculation Model for Transient Analysis

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Serpent User Group Meeting,
Helsinki, Finland, 29.05. – 01.06.2018

Work supported by the German Federal Ministry for Economic Affairs and Energy
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Motivation

- **Transient events**
  - Normal operation
    - Reactor start and shutdown, load following operation
  - Abnormal operation
    - Reactivity Initiated Accidents (RIA), Loss of coolant (LOCA), Loss of flow (LOFA)
    - Anticipated Transient without SCRAM (ATWS)

- **Reactor dynamics**
  - Neutron kinetics and thermal-hydraulics (and fuel behaviour) build a coupled system through intrinsic feedback effects
  - Necessity for coupled simulations in order to:
    - predict the reactor core behaviour adequately
    - observe if safety criteria (pin power, temperature, heat flux) are satisfied on a plant-wise, assembly-wise and pin-wise level
Motivation: Neutron Transport Methods

**Nodal Diffusion**
- Neutron transport with approximations on a coarse mesh
- Successfully applied for steady-state and transient analysis with thermal-hydraulic feedback
- Preprocessing: Lattice-physics parameters
- Detailed analysis only possible with additional measures, e.g. pin power reconstruction

**Development objective:**
A hybrid deterministic / stochastic calculation model for transient analysis

**Monte Carlo Neutron Transport**
- No major approximations in space and energy
- Successfully applied for steady-state analysis without and with thermal-hydraulic feedback
- Direct time-dependent MC still extremely challenging for large systems

Rod ejection
Control rod ejection, PWR

Power distribution
PWR UO2/MOX core
Motivation:

Why integrate Monte Carlo into transient analysis?

- Lattice-physics parameters
  - 2D fuel assembly model in an infinite lattice
  - Subregions of a 3D model
  - Kinetic parameters of the 3D model

- Output quantities
  - Multiplication factor
  - Neutron flux, power distribution on a detailed spatial level

- Further advantages
  - Continuous-energy cross section libraries
  - Complex geometries possible

Detail of PWR UO2/MOX core, various fuel assemblies, baffle and reflector
Motivation: Lattice-physics 2D / 3D

Studying the trends of lattice-physics parameters as a function of fuel temperature and moderator density of:

• 2D UO$_2$ fuel assembly model in an infinite lattice
• UO$_2$ node in the middle layer of the 3D C5G7 full-scale model
Motivation: Lattice-physics 2D / 3D (2)

Thermal fission XS and absorption XS as a function of fuel temperature

Calculations performed with Serpent 2

Conclusion

- An offset is observed between the 2D and the 3D generated homogenized cross-section → impact of the 3D neutron flux spectrum
- Both XS types (thermal) follow the same linear trend over the fuel temperature
Motivation: Lattice-physics 2D / 3D (2)

Thermal fission XS and absorption XS as a function of moderator density

Conclusion

- An offset is observed between the 2D and the 3D generated homogenized cross-section → impact of the 3D neutron flux spectrum
- Both XS types (thermal) follow the same trend over the fuel temperature, but the trend shows a curvature
Methodology: Reactor dynamic calculation

- **TH:** Thermal-hydraulic: ATHLET (GRS)
- **NK:** Neutron-kinetic: QUABOX/CUBBOX (GRS)
- **MC:** Monte Carlo: Serpent 2 (VTT Finland)
- **Mod:** Update of lattice-physics parameters

NK/TH coupling scheme (Staggered time synchronisation scheme)

NK/TH coupling scheme incl. XS updating through MC calculations
**XS Update with XS from Full-scale MC Calculation**

**First idea**

**Linear extrapolation**

- Multi-dimensional extrapolation in $T_f$, $\rho_M$, $C_B$ with absolute ranges for $\Delta T_f$, $\Delta \rho_M$, $\Delta C_B$

\[ \sigma'(T_f\,FS + \Delta T_f, \rho_M\,FS + \Delta \rho_M, C_B\,FS + \Delta C_B) = \]
\[ \sigma_{FS}(T_f\,FS, \rho_M\,FS, C_B\,FS) + \frac{\partial \sigma}{\partial T_f} \Delta T_f + \frac{\partial \sigma}{\partial \rho_M} \Delta \rho_M + \frac{\partial \sigma}{\partial C_B} \Delta C_B. \]

**Example:**

Extrapolation in 1D, only $T_f$.
Model: C5G7-TD Core

- Spec. from OECD/NEA C5G7-TD Benchmark
- 8 x UO₂ (3.7%) fuel assemblies, 8 x MOX (4.3%, 7.0%, 8.7%) fuel assemblies
- AgInCd control rods in the central four UO₂ FA
- Active length: 128.52 cm
- Conventional branching:
  - Fuel temperature [K]: 500.0, 800.0, 1000.0, 1200.0
  - Moderator density [g/cm³]: 0.62822, 0.66114, 0.71187, 0.75206
  - Boron concentration [ppm]: 500.0, 1000.0, 1500.0, 2000.0, 2500.0
Transient Analysis – Without Updating XS
Neutron Flux Distribution

Serpent calculation every 0.5 s with the current 3D distribution of TH parameters, but without feedback of XS to QC-ATHLET

Axial distribution of thermal neutron flux at 1.11 s and at 11.51 s
**Multiplication Factor – NK and Serpent**

Serpent calculation every 0.5 s with the current 3D distribution of TH parameters, but without feedback of XS to QC-ATHLET.

Maybe fixed source mode is more suitable than criticality source mode?
Update of Group Constants

Group constants of every node of the full-scale model were extrapolated with absolute ranges for fuel temperature, moderator density and boron concentration.

- Criteria will be implemented for checking extrapolation ranges in the future
- Conventional group constants were generated with 100.0E+06 neutron histories in order to obtain a smooth trend
Summary & Outlook

- Development of a concept for incorporating MC neutron transport into transient calculation scheme in order to update the cross sections (XS) during the transient
- Update of XS via an extrapolation scheme on the basis of conventional XS and XS from the full-scale MC model
- Interface implemented for data exchange between the NK / TH code system and Serpent at time steps during the transient

A first study was performed with the C5G7-TD core:
- Reactivity insertion due to rod movement
- Serpent full-scale calculation every 0.5 s
- Qualitative agreement of the evolution of the NK/TH eigenvalue and the Serpent full-scale eigenvalue
- XS updated, but have not yet been feed back to the NK code

Next steps:
- Improvement of the methodology
- Finalising the implementation of the whole calculation chain