SHARP Neutronics

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SHARP Neutronics Goal

Perform high-fidelity deterministic neutronics simulation for any reactor types with complex geometry and phenomena

- Seamless coupling with the SHARP multi-physics simulation toolkit
- Modeling flexibility for various reactors in terms of geometry and cross sections
- Reasonable computational performance to meet users’ needs
High-fidelity neutron transport code
- 2nd order discrete ordinate (SN)
- 3D MOC: a rigorous formulation with 2D MOC
  + Galerkin finite element based method in the axial direction, based on extruded geometry
- Can simulate geometric deformations

Unstructured finite element mesh with DOFs >$10^{12}$

Parallelization in space, angle, and energy
- 90% strong scaling, 75% weak scaling

Transient capability (adiabatic)
- Improved Quasi-Static (IQS) option is being developed under a NEUP project

NODAL solver option available
**Neutron cross sections**
- Resonance self-shielding with analytic Doppler broadening, ultrafine-group (~2000 groups) transport calculations
- Supports both conventional and high-fidelity codes
- Recently, updated thermal cross section library and added a 3-D MOC capability (same as PROTEUS-MOC)
- Substantial V&V tests against fast reactor benchmark problems as well as experiments including LANL, ZPPR, ZPR, BFS, Monju, EBR-II

**Gamma heating and cross sections**
- Recently extended from 21 to 94 groups

<table>
<thead>
<tr>
<th>Model</th>
<th>MC²-3 (Δk pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Unit Cell</td>
<td>1-D</td>
</tr>
<tr>
<td>Wrapper Tube</td>
<td>-163</td>
</tr>
<tr>
<td></td>
<td>-310</td>
</tr>
</tbody>
</table>
Other Cross Section Generation Options

**Cross section library**
- Is generated using NJOY and MC\(^2\)-3, based on the subgroup method or the resonance table method
- The cross section API generates cross sections inside PROTEUS on the fly
- Cross sections for thermal reactors
- Up to a few hundred groups

**Monte Carlo codes**
- Cross sections for thermal or fast reactors are generated using Serpent or OpenMC Monte Carlo codes and converted to the format that PROTEUS or conventional codes can read
Mesh Generation

- Can use any meshing tools that generate Exodus-II format
- CUBIT (developed by SNL)
  - An option for very complex geometries such as ATR
  - User must create geometry model as well as the mesh
- In-house meshing tools (User Friendly mesh)
  - Automates meshing of standard reactor configurations
    - Assembly ducts, pin cells, boundary layers
  - No CUBIT or other external software is required
  - Extrusion is the only option for 3D
Reactivity Perturbation & Sensitivity Analysis

- Spatial distribution of perturbations for a given reactor system
  - Very useful in understanding how different parts of a reactor (core, blanket, reflector) contribute to the total reactivity worth

High leakage or strong heterogeneity

- Diffusion theory shows considerable errors compared with transport results
- PERSENT provides both 3D diffusion and transport perturbation options
Applications
ABTR Simulation with PROTEUS
- Multi-group cross section generation using MC²-3
- Two models in terms of heterogeneity
  - Partially homogeneous assemblies
    (heterogeneous duct + homogeneous fuel)
  - ~1% error on control rod worth relative to MCNP
  - Less than 200 pcm error in k-effective
  - Full spatial resolution

Non-uniform structural deformation
- Is capable of detailed neutronics modeling any type of deformed geometry given from structure codes like Diablo or NUBOW
- Can be performed fully in-memory
Multi-resolution calculation with mixed local resolutions

- Model A - Homogenized assembly model (as generally considered in applications of current deterministic codes, notably DIF3D-VARIANT)
- Model B - Explicit representation of wrapper tube and inter-assembly sodium gap for all fuel regions
- Model C - Explicit pin by pin representation of a single assembly in the inner core, leaving a full material homogenization in all other assemblies
Fast Reactor (Cont’d)
Shielding

- PGSFR simulation using PROTEUS-SN for shielding calculation
  - Challenging with conventional codes to accurately estimate neutron fluxes at outside core regions
  - PROTEUS eigenvalue agrees well with MCNP within ~100 pcm for 2D problems and detailed shielding calculation is ongoing
Advanced Test Reactor (ATR)

- Complex geometry and composition assignment
  - Complex serpentine core design
  - Very narrow fuel regions
  - 93% enriched uranium in aluminum matrix

- Good agreement in eigenvalue (< 300 pcm) and flux distributions (< 4.5%) at the fuel region between PROTEUS and MCNP
Transient Reactor Test Facility (TREAT)

- Experiment performed in early TREAT operation
  - Minimum Critical Core (MinCC)
  - Complex-geometry components
- Latest, best-documented historic TREAT experiments
  - M8 power calibration experiment (M8CAL)
- IRPhEP benchmark problems

<table>
<thead>
<tr>
<th>Core</th>
<th>Case</th>
<th>MCNP or Serpent</th>
<th>PROTEUS (Δk, pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>MinCC</td>
<td>2D partial core</td>
<td>1.29939 (±15)</td>
<td>-167</td>
</tr>
<tr>
<td></td>
<td>3D core</td>
<td>1.00490 (±19)</td>
<td>115</td>
</tr>
<tr>
<td>M8CAL</td>
<td>3D partial core</td>
<td>1.37609 (±16)</td>
<td>147</td>
</tr>
<tr>
<td></td>
<td>3D core *</td>
<td>1.00497 (±18)</td>
<td>148</td>
</tr>
</tbody>
</table>

* simplified model
The only university research reactor in the US to use fuel rods similar to operating commercial LWRs

- Generated meshes using CUBIT + UFmesh
- Excellent agreement in eigenvalue between PROTEUS-MOC and Serpent

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<th>MCNP or Serpent</th>
<th>PROTEUS (Δk, pcm)</th>
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</thead>
<tbody>
<tr>
<td>RCF</td>
<td>2D partial core</td>
<td>1.26661 (±9)</td>
<td>-4</td>
</tr>
<tr>
<td></td>
<td>3D core</td>
<td>0.99337 (±10)</td>
<td>24</td>
</tr>
</tbody>
</table>
PROTEUS-MOC is able to provide accurate solutions for neutron streaming through large CR holes.

Preliminary calculations on 3D fuel assembly problems indicated good agreement (< 90 pcm) with Monte Carlo solutions without introducing any methodology patches.
Light Water Reactor

**C5G7 PWR Benchmark**

<table>
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<th>PROTEUS</th>
<th>Δk, pcm</th>
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</thead>
<tbody>
<tr>
<td>Unrodded</td>
<td>1.14308 (3)</td>
<td>1.14310</td>
<td>2</td>
</tr>
<tr>
<td>Rodded A</td>
<td>1.12821 (3)</td>
<td>1.12817</td>
<td>-4</td>
</tr>
<tr>
<td>Rodded B</td>
<td>1.07777 (3)</td>
<td>1.07750</td>
<td>-27</td>
</tr>
</tbody>
</table>

- Pin power error in the unrodded case: max 0.9%, RMS 0.2%
Argonne Fast Reactor Codes

- **ENEF/B** (Evaluated Nuclear Data Files)
- **MC²-3** (Slowing Down Solver)
- **Multi-group Cross Sections**
- **DIF3D/REBUS-3** (Flux Solver and Fuel Cycle Performance Analysis)
- **Power Distribution, Depletion Data, etc.**
- **SE2-ANL** (Steady-State Thermal Hydraulics)
- **Temperature Margins, Flow Distributions**
- **SAS4A/SASSYS-1** (Transient Safety Analysis)
- **Whole Plant Transient Responses**
- **Temporal Margins, Flow Distributions**
- **PERSENT** (Perturbation Theory)
- **Reactivity Feedback and Kinetic Data**
- **Plant Design Information**
- **DIF3D-VARIANT**

**Fuel Management Strategy**

**Reactor Design Parameters**

**Transient Scenarios**

**Plant Design Information**
Stability questions
- Impact of coolant density change during core transit
- Most designs consider activated fuel leaving the core
- Loss of flow leads to positive feedback in the core
- Impact of multiple flow paths (blanket/core) on control system

Updated a version of DIF3D to explore the stability problems associated with moving fuel
- Allows multiple coolant flow channels through reactor
- Tracks precursor distribution in-core and ex-core
- Different channel time delays for reprocessing bleed

Analysis showed that fuel cycle behavior is not impacted by flowing fuel behavior
- $k_{\text{eff}}$ can drop by 200 pcm depending upon flow
- Significant radiation source in out-flow reflector/shielding and ex-core piping
Validation Database

**ZPPR-15 experiments**
- Doppler measurement
- Axial expansion measurements
- Foil measurement
- Neutron spectrum measurements
- Gamma dose measurements
- B-10 reaction rate measurements
- Control rod and sodium void worth measurements

**BFS experiments (I-NERI with KAERI)**
- Control rod worth, sodium void worth, aluminum rod worth, axial and radial expansion measurements

**EBR-II experiments**
- Core follow for 10 years (1984 – 1994)
- Depletion data
Summary

- **NEAMS neutronics tools**
  - Neutronics transport code: PROTEUS (SN, MOC, NODAL)
  - Cross section generation tools: MC$^2$-3, Cross section API, Monte Carlo
  - mesh generation tool: UFmesh
  - Perturbation and sensitivity analysis tool: PERSENT
  - Software development QA: BuildBot

- **V&V tests**
  - Fast reactors (ZPPR, ABTR) and various thermal reactors (ATR, TREAT, PWR (C5), RPI research reactor, VHTR, etc.)

- **Improved ANL code suite**
  - MC$^2$-3, DIF3D/REBUS, PERSENT
  - Substantial V&V practices against ZPPR-15, EBR-II, etc.
  - Being used by ART, TerraPower, KAERI for actual fast reactor design
Software Status

Software QA
- All codes are under the SVN version control, tracking source code changes and impacts on verification test suite
- Nightly regression tests using BuildBot (http://buildbot.net/) to ensure continued accuracy and performance

Software availability / deployment / licensing
- All physics codes are export controlled (licensing required; free for government use)
- ANL TDC personnel supports for code licensing
  - Elizabeth K. Jordan (ekjordan@anl.gov) at the TDC division or nera-software@anl.gov

Required computational resources
- PROTEUS requires parallel machines with 500 – a few tens K processors
- All other codes can run in a serial mode on a regular Linux machine

Training upon request
- Methodology / user manuals and training material are available

Contact: nera-software@anl.gov
Questions?

- **PROTEUS Users**
  ART, CESAR, ORNL, INL, RPI, Purdue, Florida, Penn State, UM, KSU, NCSU, UMass-Lowell, Rnet-tech

- **MC²-3 Users**
  ART, TerraPower, ORNL, INL, BNL, Berkley, MIT, Purdue, Georgia Tech, Tennessee, NCSU, Florida, (Korea) KAERI, UNIST, SNU

- **PERSEENT Users**
  ART, KAERI