Nuclear Analysis Capabilities for Fusion Energy Systems

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Nuclear Fusion

Unlike fission where uranium splits generating energy, fusion occurs when two hydrogen nuclei fuse together and release energy

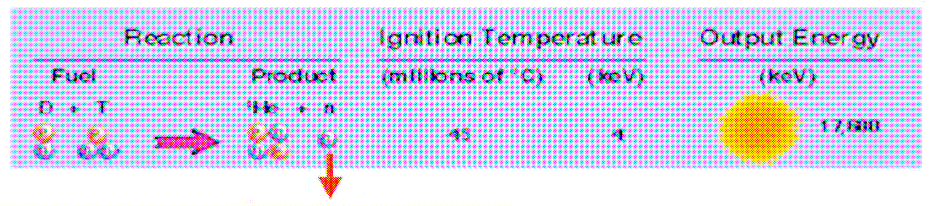
Two approaches:

- Magnetic confinement
- Inertial confinement

D-T Fusion Represents a Nearly Inexhaustible Energy Source

Fuels: **Deuterium**: abundant in sea water

Tritium: Half-life~12 years…must be produced?



^{6}Li + n --> T + ^{4}He + 4.8 MeV + others

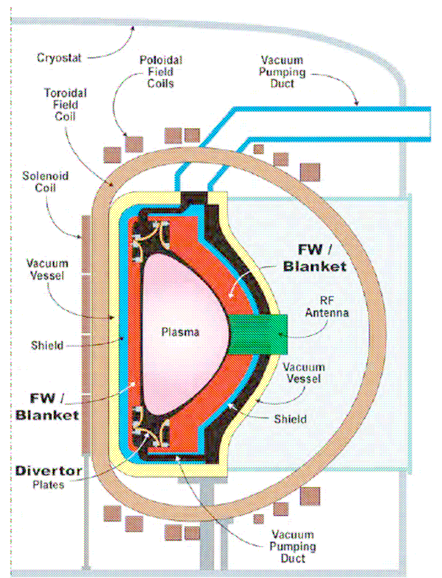
$T_{bred} / T_{burned} > 1$

"Real" fusion fuel cycle:

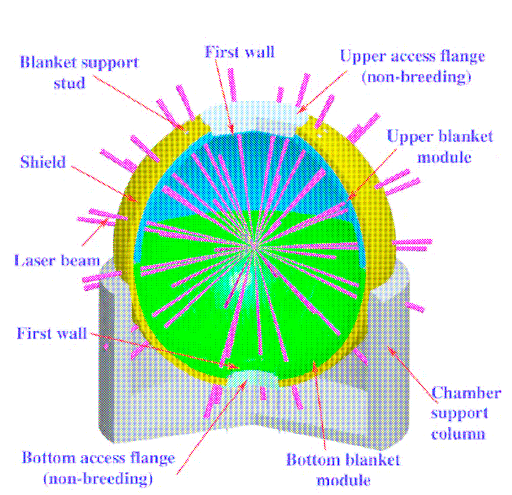
^{6}Li + D = 2^{4}He + 22.4 MeV

30 Million years of world energy demand in oceans

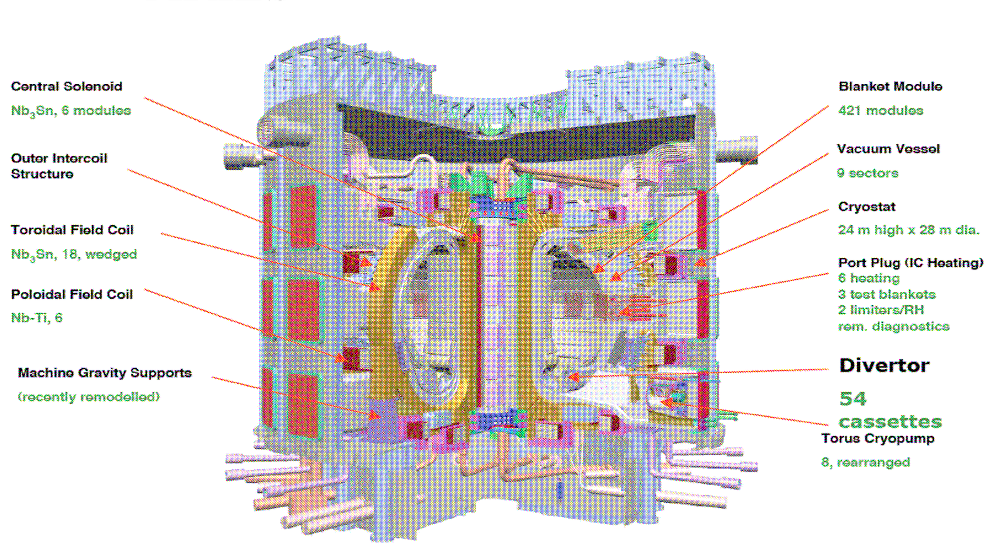
Fusion Reactors are Complex with Many Components



ARIES



HAPL



ITER

1st Integrated Fusion Test Reactor

- Agreement signed November 21, 2006
- ITER construction starts in 2007 at Cadarache, France.
- First plasma in 2016 and 20 years of operation.

Nuclear Analysis

Nuclear Analysis is Essential Part of Design

- Energetic 14 MeV neutrons are emitted in plasma and slowed down and absorbed by surrounding components
- Nuclear analysis for components surrounding the plasma is essential element of fusion nuclear technology
  - Tritium production in breeding blankets to ensure tritium self-sufficiency
  - Nuclear heating (energy deposition) for thermal analysis and cooling requirement
  - Radiation damage in structural material and other sensitive components for lifetime assessment
  - Provide adequate shielding for components (e.g., magnets) and personnel access
  - Activation analysis for safety assessment and radwaste management
- State-of-the-art predictive capabilities (codes and data) are needed to perform required nuclear analyses

Calculation Methods

- There are several numerical methods and codes available to solve the Boltzmann transport equation
- The methods can be broken down into two groups
  - *Deterministic method*: Directly solves the equation using numerical techniques for solving a system of ordinary and partial differential equations
  - *Statistical based method*: Solves the equation using probabilistic and statistical techniques
- Each method has its strengths and weaknesses

Deterministic Approach

The phase space (space, angle, energy) is discretized

Angle

S_n - Discrete Ordinates P_l - Moment expansion

Energy

Multi-group (175n-42g)

Spatial discretization

Finite Element (un-structured meshes)

Finite Difference (structured equal fine meshes)

Advantages

- Spatial Resolution
- Flux evaluated at a large number of points

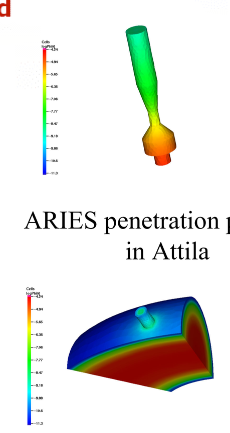
Disadvantages

- Angular Quadrature approximation
- Legendre Polynominal expansion of cross- sections
- Ray-Effects for streaming problem
- Group treatment of energy variable
- Require large storage space for multi-dimensional calculations

Codes

DANTSYS, DOORS. PARTISN code systems (1D, 2D, 3D finite difference)

ATTILA (3D finite element with CAD coupling)



ARIES penetration problem in Attila

Monte Carlo Approach

Method

- Use probabilistic and statistical approach to solve transport equation
- The particle travel distance and interaction physics are converted to probabilistic and cumulative distributions, that are sampled using a random number

Advantages

- Exact Geometrical representation
- Exact treatment of the transport process
- Exact source-modeling capability
- Continuous (pointwise) energy treatment of the cross-sections

Disadvantages

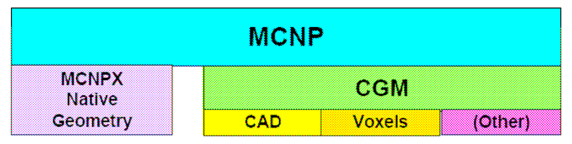
- Require variance reduction techniques to improve accuracy
- Usually cannot generate accurate results at all locations
- Many particle histories and large CPU time to obtain accurate results

Codes

MCNP, MCNPX, MORSE, TRIPOLI, TART

Developed Innovative Monte Carlo Tool MCNP-CGM

- Direct use of solid model geometry in MCNP
  - Use Common Geometry Module (CGM) to interface MCNP *directly* to CAD & other geometry data



- Production experience
  - ARIES-CS
  - HAPL
  - ITER FWS

Motivations

- Reduce impacts of manual conversion of 3-D model data
  - Time
  - Simplifications
  - Errors
- Extend richness of geometric representation

Activation Codes

Method

Solve rate equations for radioactive nuclide production and decay to determine radioactive inventory, decay heat, biological dose, and radwaste

Codes

ALARA
DKR-PULSAR
REAC2
RACC
FISPACT
ORIGN2
ANITA
ACAB
ACT4

Nuclear Data

Evaluated nuclear data include raw data that needs processing to produce working libraries for use with nuclear analysis codes

US: ENDF/B-IV, -V, -VI, -VI

ENDF/B-VII released Dec 15, 2006

Japan: JENDL-3.2, JENDL-3.3

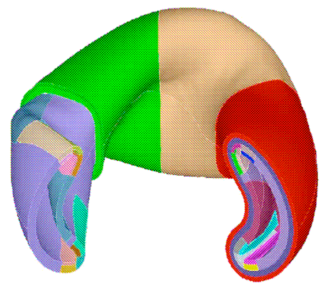
EU: JEFF-3.1

RF: BROND-2.1

Processing Codes: NJOY, TRANSX, AMPX

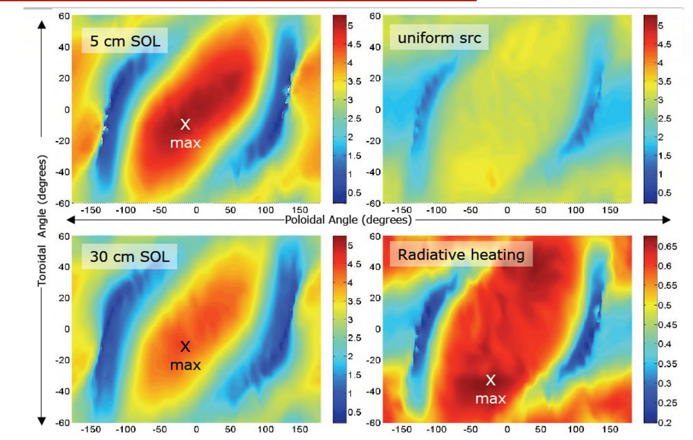
- Process data in either Multi-group or continuous energy format
- In addition to basic transport and scattering cross sections, special reaction cross sections are generated
  - Kerma factors- for nuclear energy deposition
  - Damage energy cross sections- for atomic displacement (dpa)
  - Gas production (tritium, helium, hydrogen)

Application to ARIES-CS Compact Stellarator



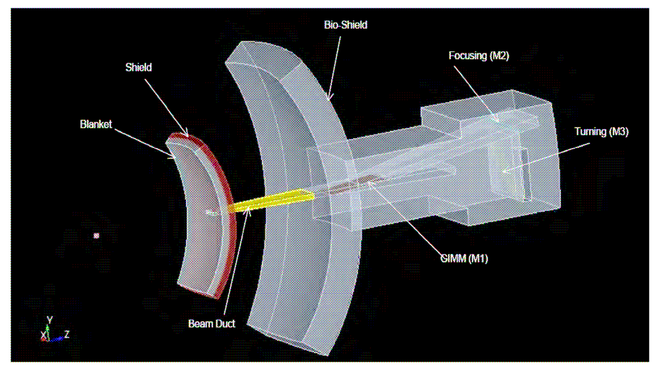
- Geometry complex varying in both poloidal and toroidal directions
- Cannot be modeled by standard MCNP

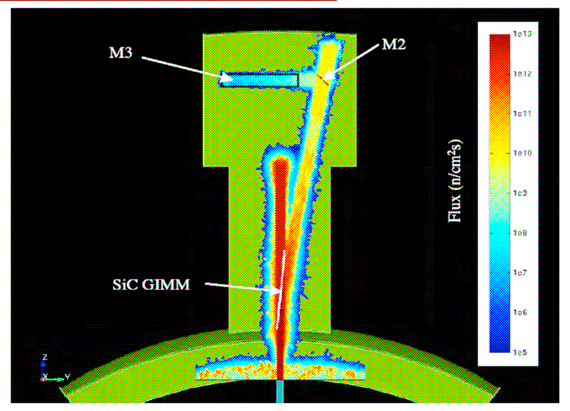
Examined effect of helical geometry and non-uniform blanket and divertor on total TBR and nuclear heating



NWL Maps (colormaps in MW/m^2)

HAPL Final Laser Optics

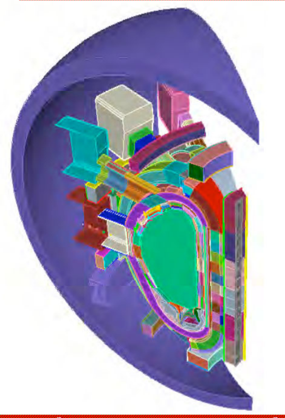




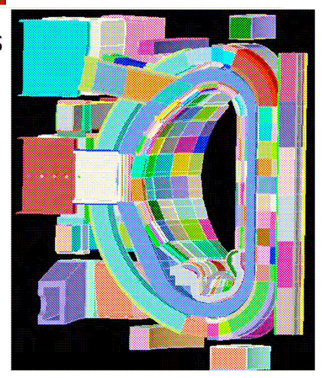
Neutron Flux in Laser Beam Duct

ITER Benchmark

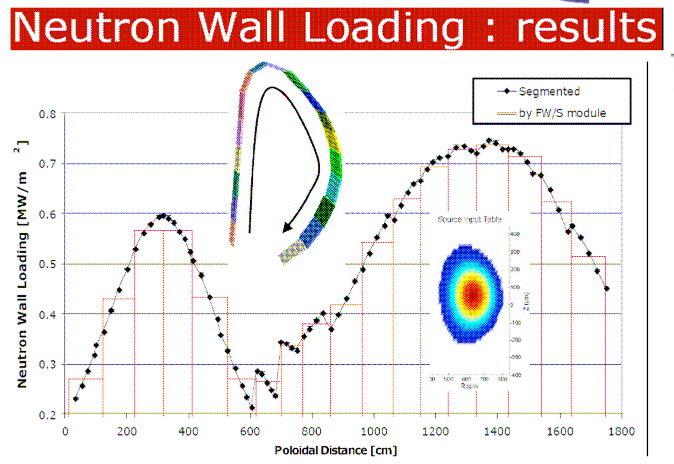
- 40 degree machine sector
- Used for validation of MCNP/CAD tool
- 802 cells
- 9834 surfaces
- 17 material specifications



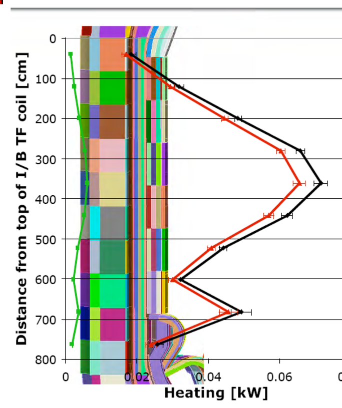
- Comparing 4 results
  - Neutron wall loading
  - Divertor fluxes and heating
  - Magnet heating
  - Midplane port shielding/streaming
- Participants
  - UW, FZK, ASIPP, JAEA



Neutron Wall Loading : results



TF Coils : results



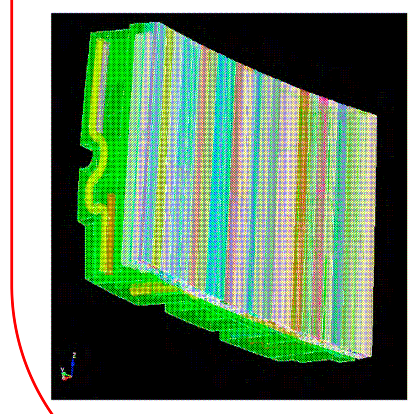
Nuclear Heating (W)

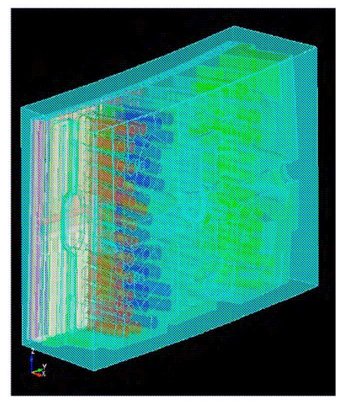
Neutron	Photon	Total
1.39 ± 0.05	17.0 ± 0.6	18.4 ± 0.6
2.47 ± 0.06	29.4 ± 0.6	31.8 ± 0.7
3.82 ± 0.04	44.6 ± 0.4	48.4 ± 0.5
5.41 ± 0.05	60.4 ± 0.6	65.8 ± 0.6
6.03 ± 0.12	65.6 ± 0.9	71.6 ± 1.0
5.16 ± 0.08	57.0 ± 0.7	62.2 ± 0.8
3.38 ± 0.04	40.9 ± 0.5	44.3 ± 0.6
2.27 ± 0.04	29.9 ± 0.5	32.2 ± 0.6
3.66 ± 0.08	45.7 ± 1.3	49.4 ± 1.4
1.88 ± 0.05	24.0 ± 0.7	25.9 ± 0.7
35.5 ± 0.2	**415 ± 2.3**	**450 ± 2.5**

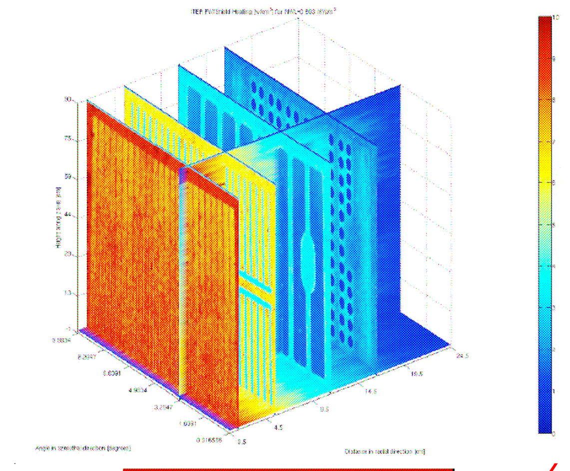
8.1 kW in all TF I/B legs

ITER First Wall/Shield Module 13 Mockup

Model generated by designers using common tools facilitates analysis







FWS results