



AECL EACL

RFSP Model



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du Canada limitée



Static Two-Energy-Group Neutron Diffusion Equation for Eigenvalue Problems

Two-group equation solved by RFSP:

$$-\nabla \cdot D_1(\vec{r}) \nabla \Phi_1(\vec{r}) + (\Sigma_{a1}(\vec{r}) + \Sigma_{12}(\vec{r})) \Phi_1(\vec{r}) - \Sigma_{21}(\vec{r}) \Phi_2(\vec{r}) - \frac{1}{k_{\text{eff}}} (v \Sigma_{f1}(\vec{r}) \Phi_1(\vec{r}) + v \Sigma_{f2}(\vec{r}) \Phi_2(\vec{r})) = 0$$

$$-\nabla \cdot D_2(\vec{r}) \nabla \Phi_2(\vec{r}) + (\Sigma_{a2}(\vec{r}) + \Sigma_{21}(\vec{r})) \Phi_2(\vec{r}) - \Sigma_{12}(\vec{r}) \Phi_1(\vec{r}) = 0$$



Static Two-Energy-Group Neutron Diffusion Equation for Eigenvalue Problems (con't)

where, at position \vec{r}

$\Phi_1(\vec{r})$ and $\Phi_2(\vec{r})$ are the group-1 and group-2 neutron fluxes respectively,

$\Sigma_{a1}(\vec{r})$ is the group-1 (fast) absorption cross section,

$\Sigma_{a2}(\vec{r})$ is the group-2 (thermal) absorption cross section,

$\nu\Sigma_{f1}(\vec{r})$ is the production cross section for the fast group,

$\nu\Sigma_{f2}(\vec{r})$ is the production cross section for the thermal

group,

$\Sigma_{12}(\vec{r})$ is the cross section for transfer from the fast group to the thermal group (also called downscatter),

$\Sigma_{21}(\vec{r})$ is the cross section for transfer from the thermal group to the fast group (also called upscatter),

$D_1(\vec{r})$ is the group-1 (fast) diffusion coefficient,

$D_2(\vec{r})$ is the group-2 (thermal) diffusion coefficient, and

k_{eff} is the reactor multiplication constant and is the

inverse of

the eigenvalue



Static Two-Energy-Group Neutron Diffusion Equation for Eigenvalue Problems (con't)

The diffusion coefficients D_g are related to the transport cross sections $\Sigma_{tr,g}$ ($g = 1, 2$) by:

$$D_g(\vec{r}) = \frac{1}{3\Sigma_{tr,g}(\vec{r})} \quad g=1,2$$

where $\Sigma_{tr,g}(\vec{r}) = \Sigma_{12}(\vec{r})(1-\mu_{o,g})$ and $(1-\mu_{o,g})$ is the correction to the scattering cross-section to take account of non-isotropic scattering.



Data Prerequisite to Solve Diffusion Equation

- ***What*** is in the reactor
- ***Where*** it is
- ***In what state*** it is (e.g., *irradiation*)

Geometry

Length and radius of calandria, radius of notch

Reflector thickness

Lattice pitch, bundle length

Mesh lines used in finite-difference model

Distinction between *mesh array* (for flux calculation) and *lattice array* (defining fuel bundles)

Axial and radial extrapolation lengths

This data entered via *DATA GEOMETRY cards.

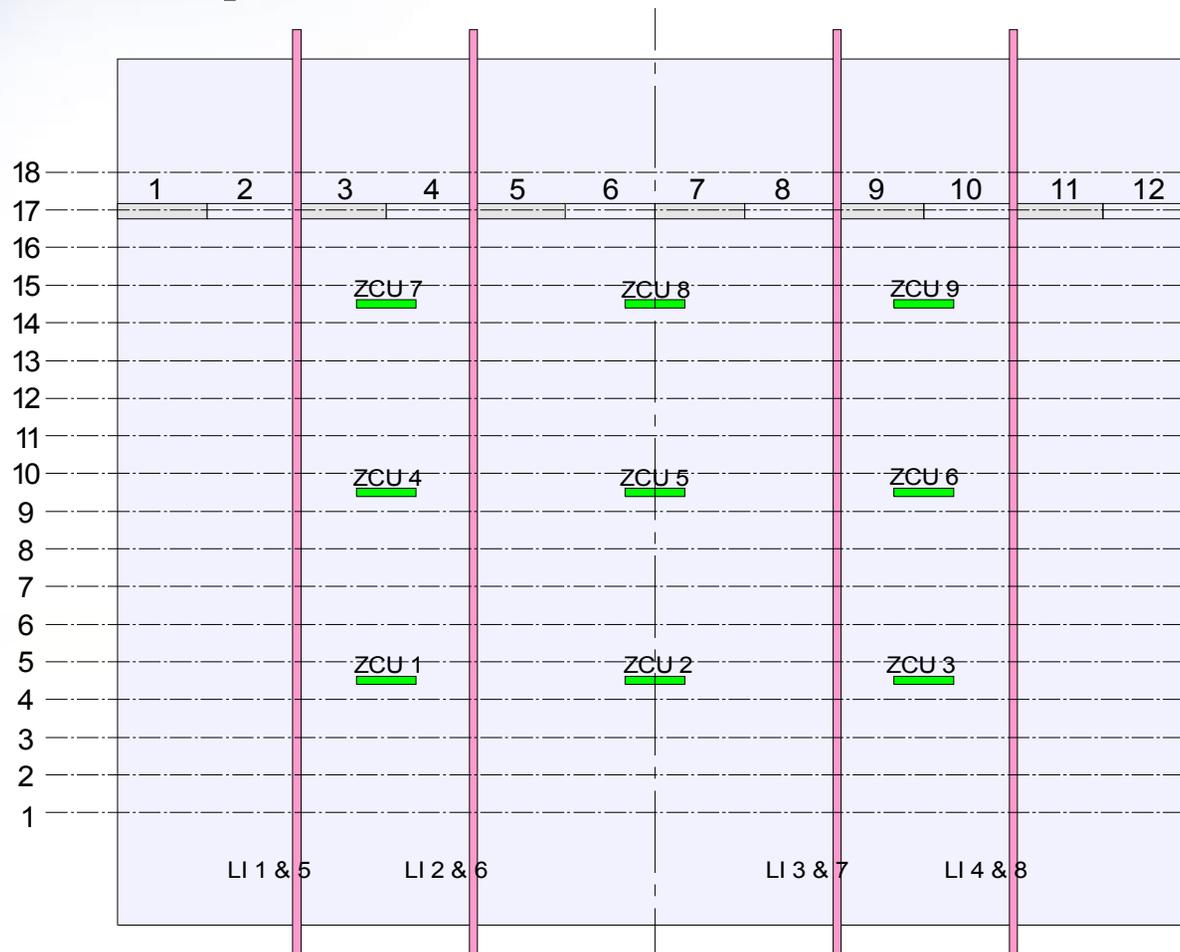


Mesh Spacing

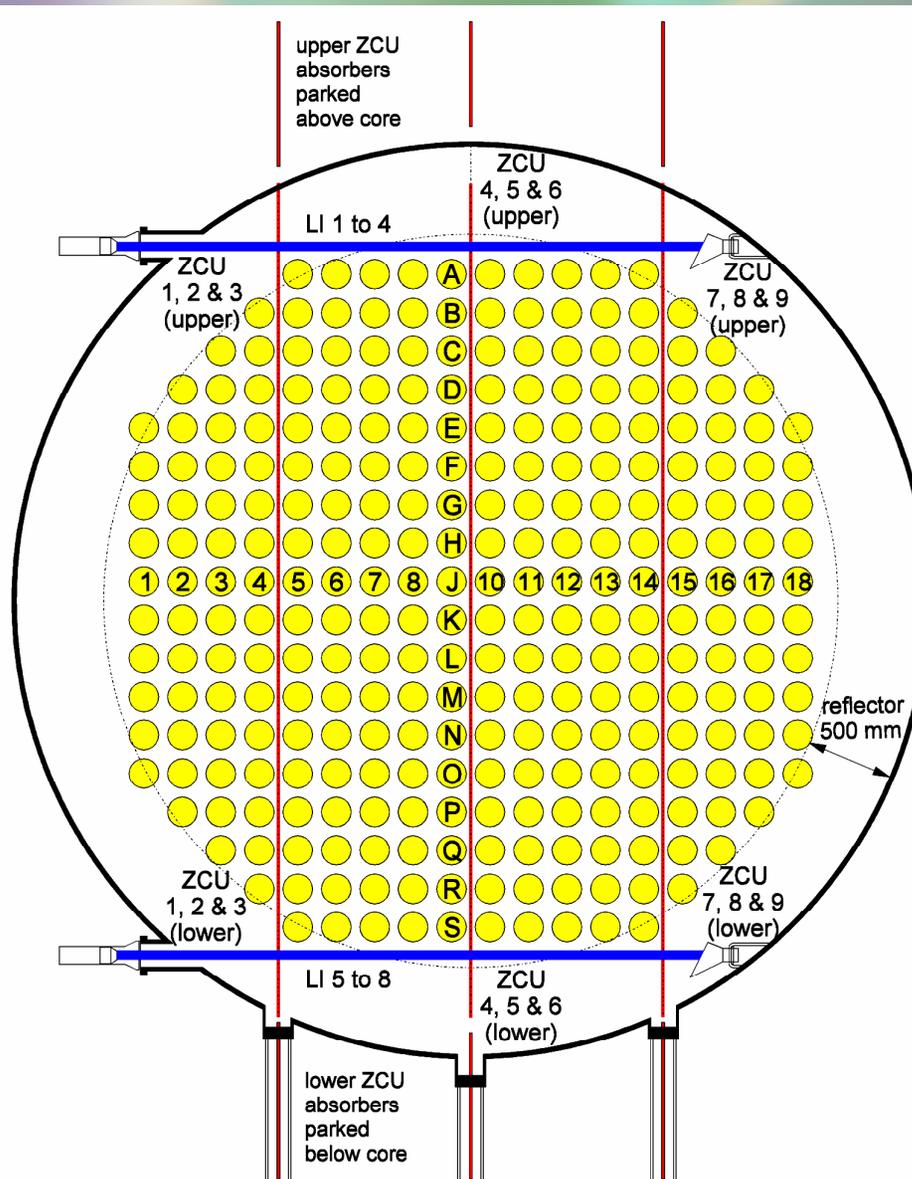
- The finite-difference model is a collection of rectangular parallelepipeds, defined by the *mesh lines*. The RFSP user defines the mesh lines in the model.
- By definition, the nuclear properties are *uniform* over each parallelepiped (and the flux is calculated at its mid-point).
- However, the properties are not *directly* defined over the parallelepipeds. They are defined over:
 - lattice cells (for lattice properties)
 - defined volume of influence (for reactivity devices)



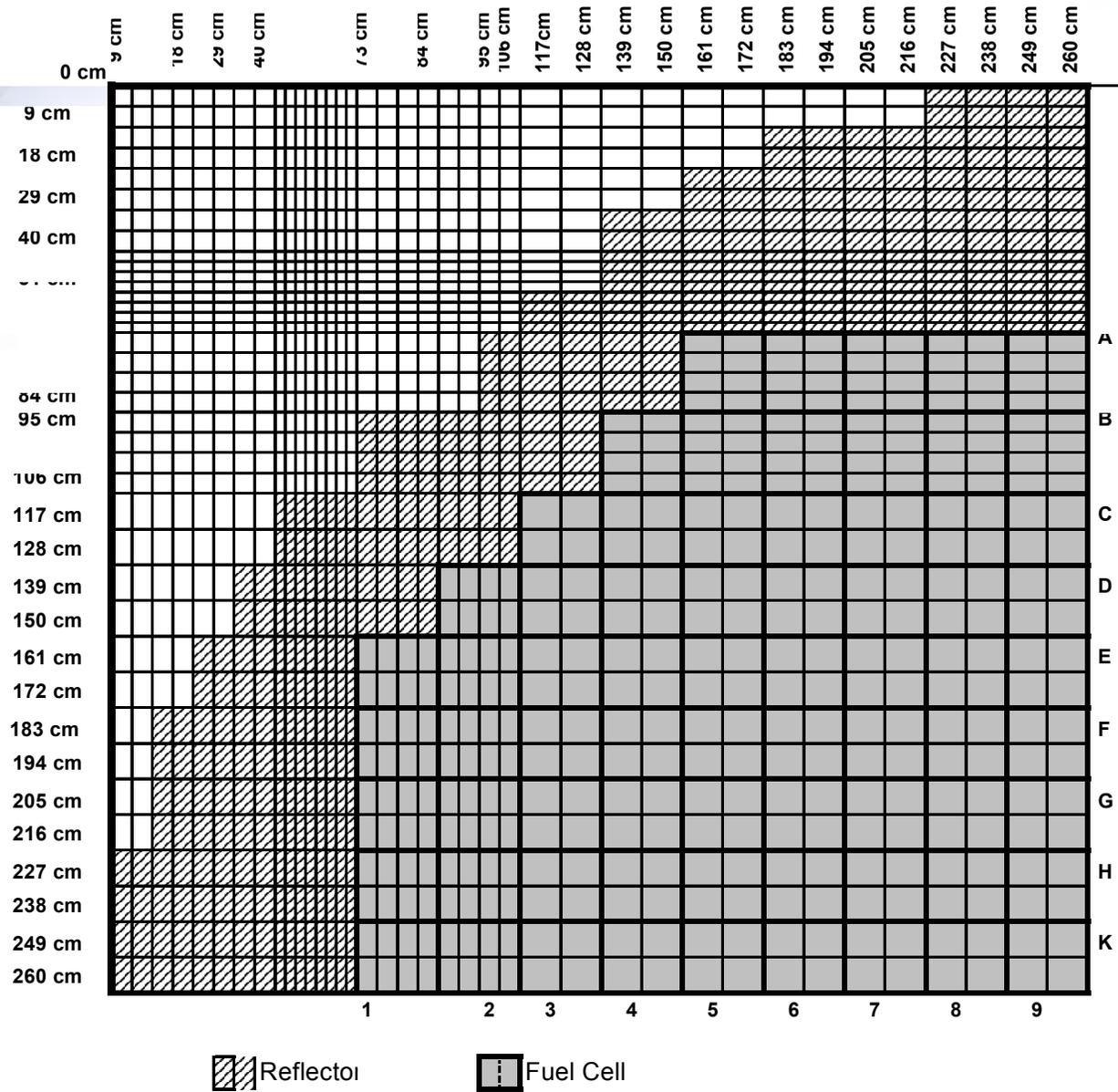
Top View of ACR Model

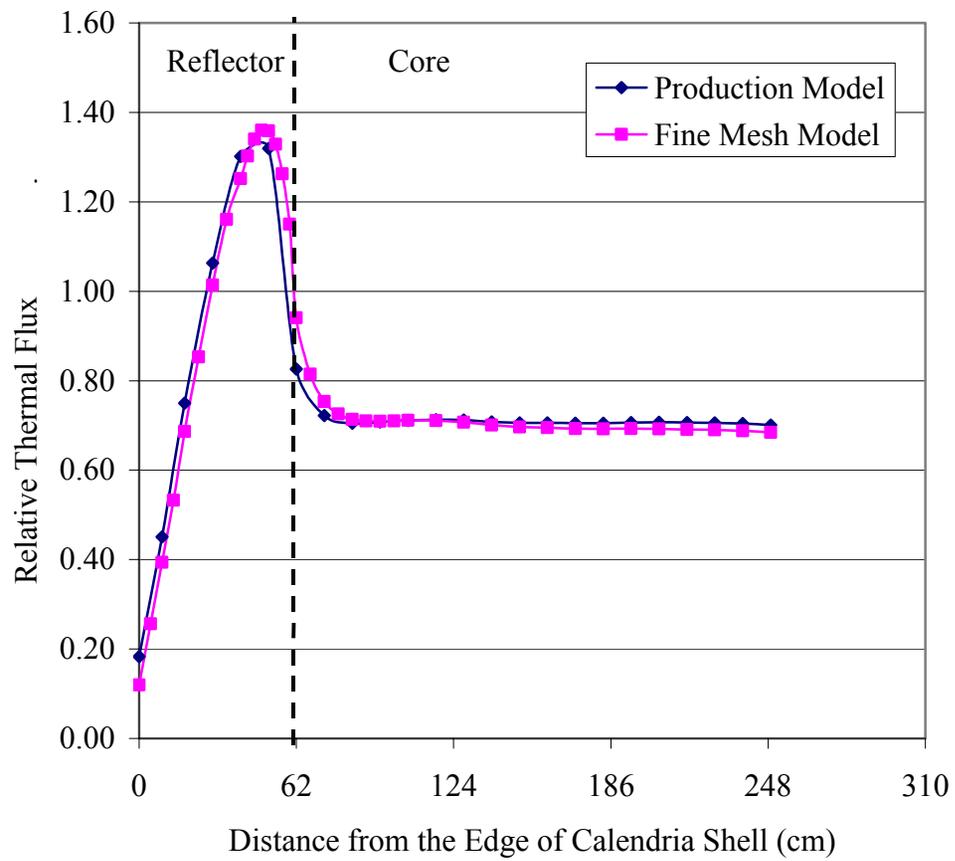


ZCU= Zone Control Unit
LI = Liquid Poison Injection Tubes



ACR-700 Reactor Core
284 Fuel Channels
220 mm Lattice Pitch







Lattice Properties Overview

- **Uniform Parameter – Simplest Method**
 - lattice properties are tabulated as functions of fuel burnup (or irradiation) only
 - assume effective core-average conditions for each fuel bundle
 - cannot account for local-parameter effects such as distributed power, xenon, coolant temperature, etc.
 - not suitable for accurate core-tracking and xenon transient simulations
 - Used for time-average and some static calculations (*TIME-AVER, *INSTANTAN and sometimes *SIMULATE)



Lattice Properties Overview Cont.....

- **Grid Based or Macro-Depletion Method**
 - lattice properties are specified as functions of fuel burnup (or irradiation) and of other local parameters
 - macroscopic cross sections are generated beforehand at various assumed operating conditions
 - Interpolation is performed for each fuel bundle to generate local-parameter-dependent macroscopic cross sections
 - does not take into account the dependence of the cross sections on a bundle's history
 - Used in kinetics module of RFSP (*CERBERUS)



Lattice Properties Overview Cont.....

- **Micro-Depletion Method**
 - as simple as the macro-depletion model
 - but addresses both:
 - the microscopic cross section of a nuclide (independent of history) and
 - its number density, which depends on the depletion history
 - Used in the *SIMULATE module for core tracking calculations



Lattice Properties Overview Cont.....

- **Direct use of WIMS – Most Rigorous Method**
 - history-based local-parameter method
 - lattice code is coupled directly with the core-analysis code, and the lattice calculations are performed for each bundle at each time step
 - impractical for routine calculations at this time
 - Some work has been performed to demonstrate feasibility of parallel processing techniques using PVM or MPI.



Reactivity Devices

- **Device properties given as incremental cross-sections over some volume of influence (usually lattice pitch x length x bundle length)**
- **Calculated by supercell code (e.g., DRAGON) and added to lattice cross sections over volume of influence**



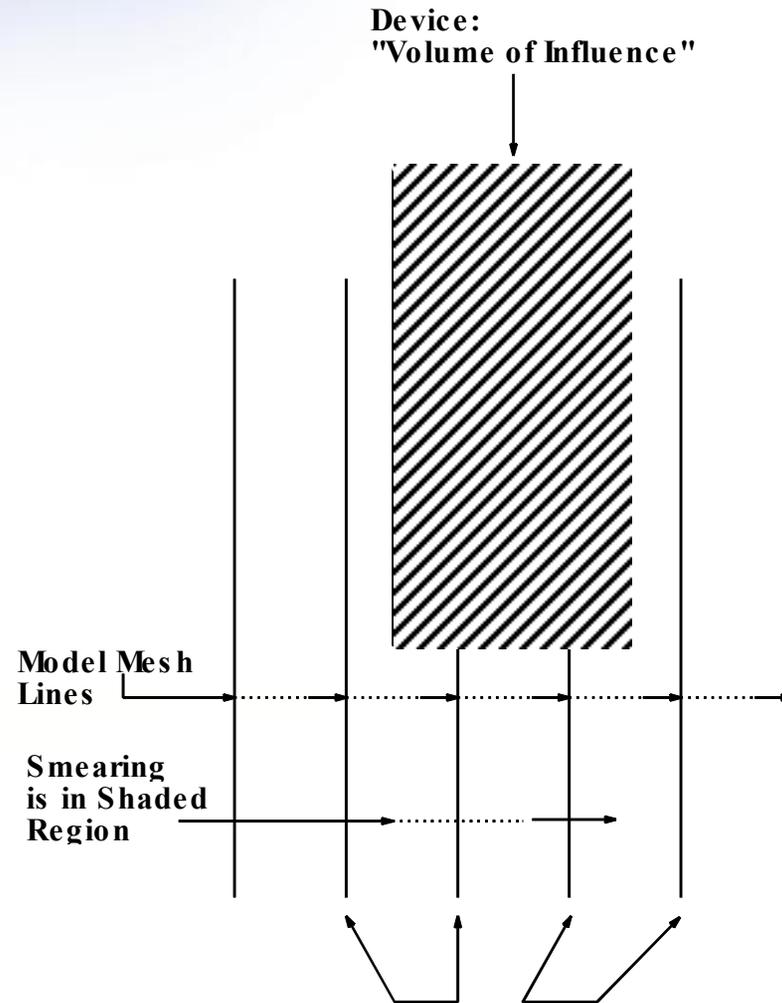
Property Smearing

- If the boundaries of the lattice cells and/or of the device volumes of influence do not coincide with mesh lines, the properties will be smeared to the *next* (outwardly) mesh line. The smearing involves “diluting” the cross-sections in inverse proportion to volume.

Smearing is not recommended because:

- It denies the volume over which the cell code or device code “wanted” to act especially in terms of absorption, a wider volume of action increases the absorption (even in the presence of dilution); e.g., reactivity worth of shutoff rods is artificially enhanced - non conservative.

Moral of Story: ensure your model has no smearing



Mesh Lines to Which Device Properties are Smeared