Chulalongkorn University
Bangkok, Thailand

CANDU Fuel Management and Advanced Fuel Cycles:

Research at École Polytechnique

by

D. Rozon

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Institut de génie nucléaire
École Polytechnique
Montréal, Qc
Canada H3C 3A7

Introduction

- Fuel Management (F/M) is essential for continued normal operation
  - changes in fuel composition with irradiation
    - fuel depletion leads to decline in potential reactivity of fuel
    - power distribution varies (slowly) with time, because burnup is not uniform
  - size of reactor core is minimized, to minimize capital costs (heavy water).
    - in a PWR, size of the pressure vessel is a similar issue
    - flux shape must be flattened to minimize peak fuel rating

- F/M objectives
  - select refueling to allow continued full power operation at minimal operating cost (fuelling costs)
    - in a PWR, batch refuelling with reactor shutdown (annual);
    - in CANDU, on-power refueling (daily)
  - satisfy safety margins
    - maintain power distribution close to nominal;
    - in CANDU, avoid ROPT trip

- F/M variables

	CANDU	***PWR***
fuel element	bundle (50 cm)	assembly (6 m)
fresh fuel	Nat U, SEU, MOX, DUPIC	SEU (3-4%), MOX
axial F/M	axial shuffling scheme	no (burnable poisons)
radial F/M	channel selection	assembly repositionning
excess reactivity	small	large (boron)

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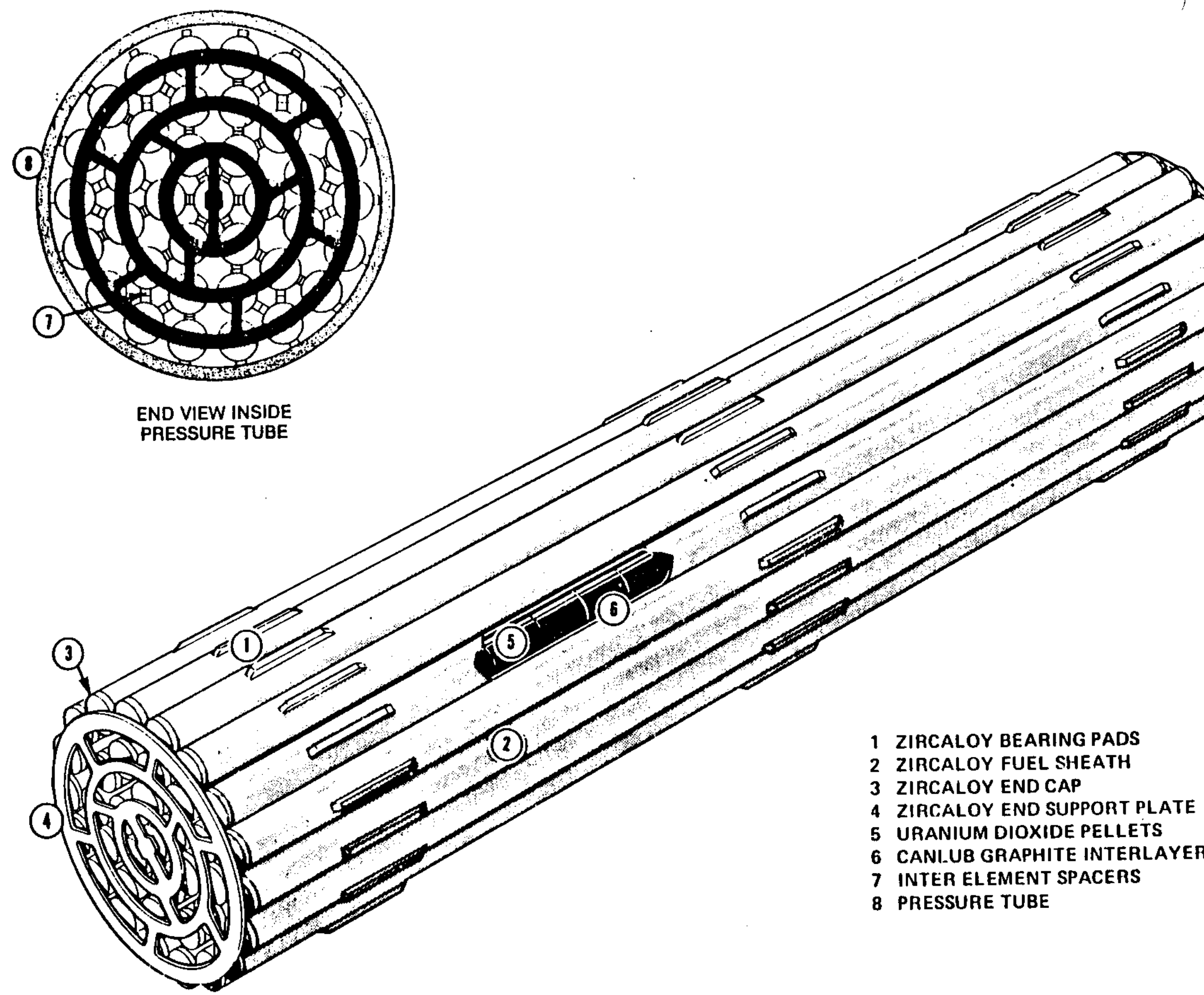


FIGURE 2.6-1 37-ELEMENT FUEL BUNDLE

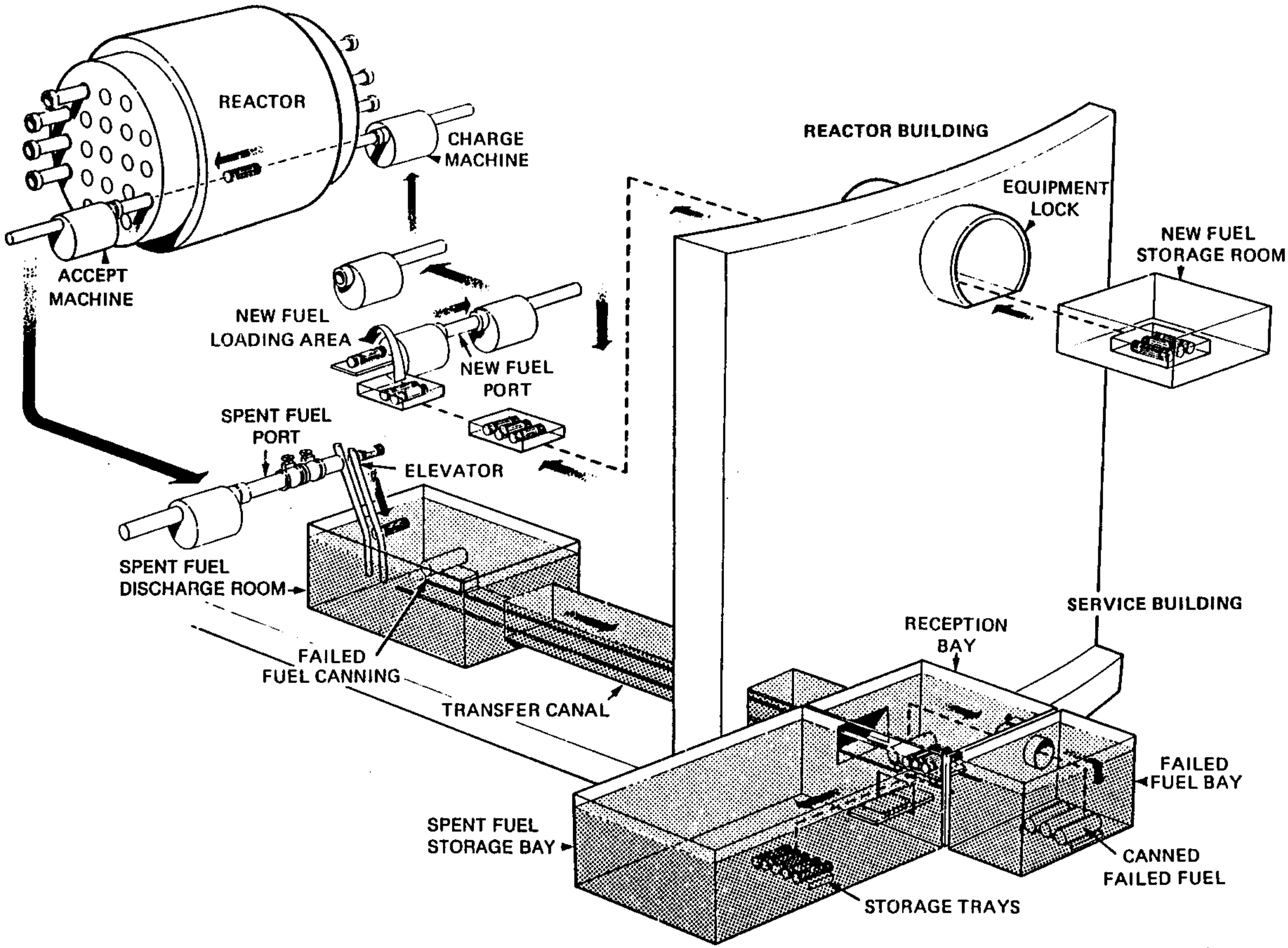


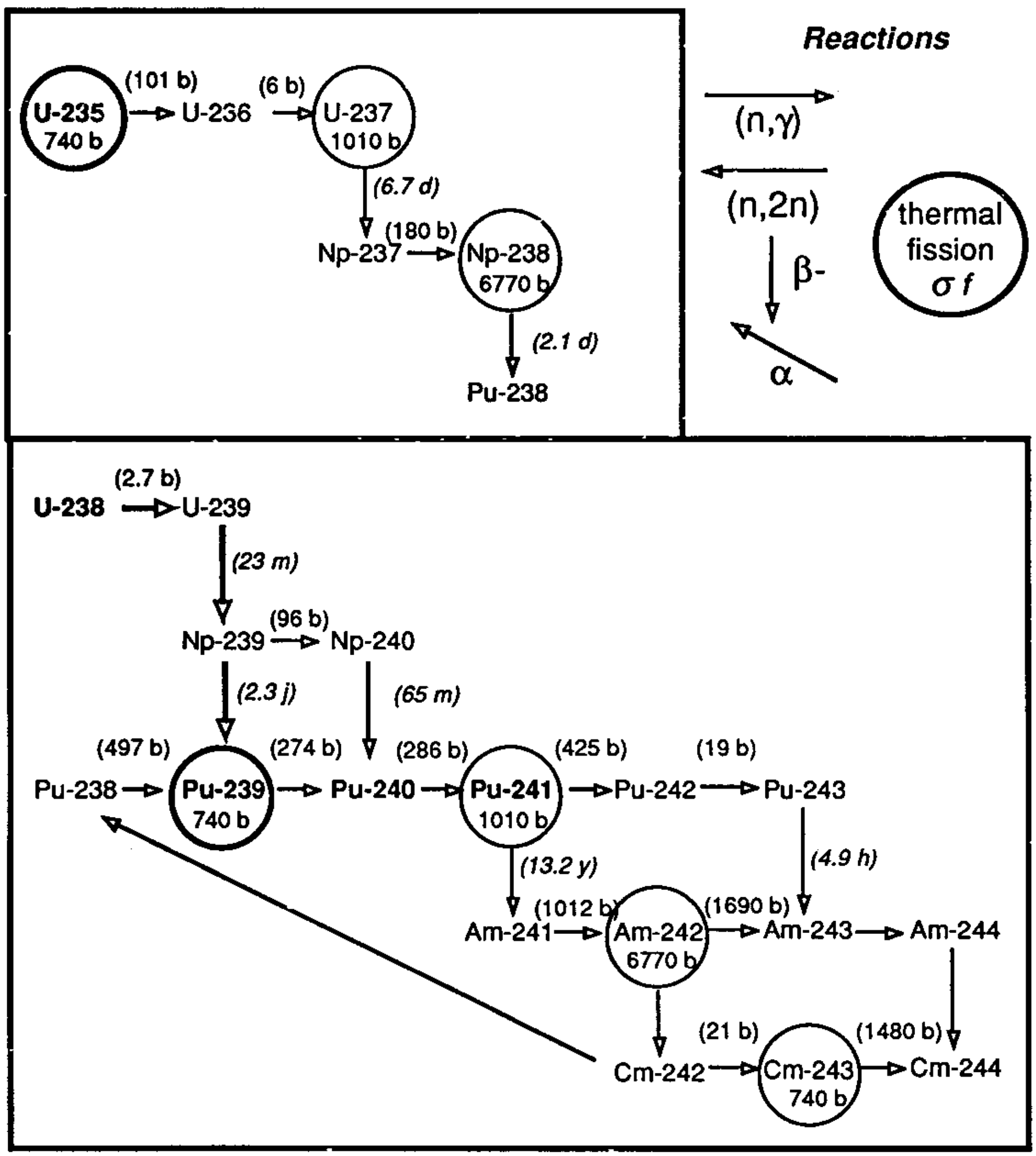
FIGURE 5.0-3 FUEL HANDLING SYSTEM SCHEMATIC

Fuel Management Research Topics

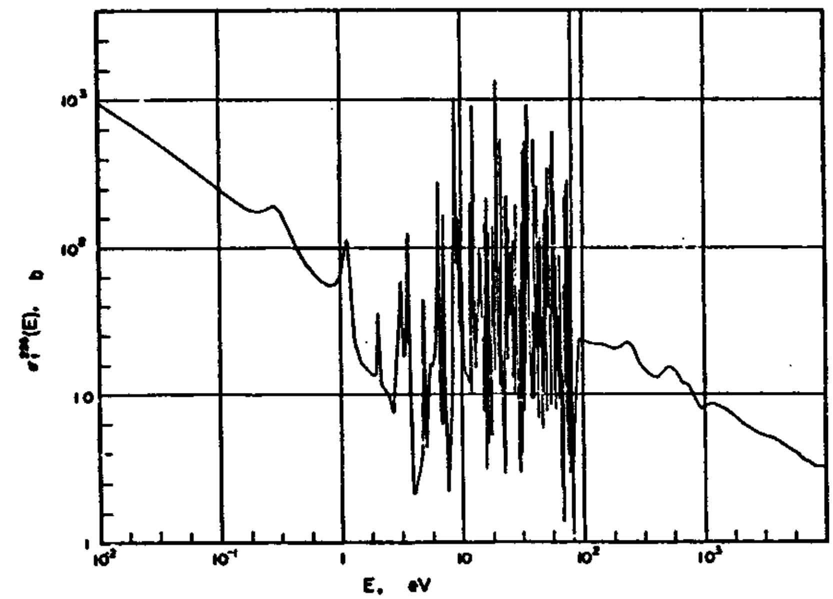
1. Computational Reactor Physics
    - DRAGON
    - DONJON
2. Generalized Perturbation Theory Applications
    - core design (equilibrium refuelling)
    - zone controller response to refuellings
3. Global Depletion Schemes
    - feedback model for local parameters
    - block depletion
4. Advanced Fuel Cycles in CANDU
    - SEU, MOX vs. Natural Uranium
    - DUPIC

1. Basic Concepts

Major Nuclear Reactions in Uranium-based Fuel



Cross Sections



- Cross sections vary with neutron energy

  - Complex behaviour
  - Evaluated Nuclear Data Files (ENDF-B-5,...)

- Neutron energy spectrum:

  - fission neutrons average 1.2 MeV

  - thermal reactors (moderator): 20°C ⇒ 0.0625 eV

    - CANDU: cold heavy water moderator (90% of volume), warm heavy water coolant in pressure tube (5% of volume)
    - PWR: warm light water moderator (coolant), occupies 65% of volume.

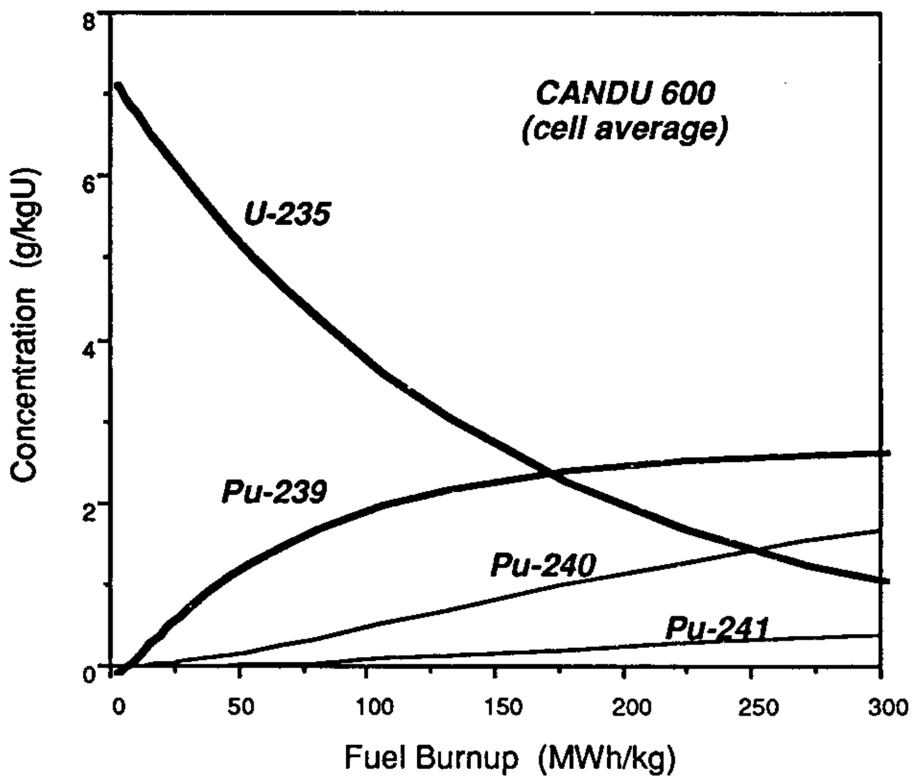
- Neutron flux distribution:

  - chain of reactor physics calculations required to determine the flux distribution (transport theory ⇒ diffusion theory)

  - reactor is highly heterogeneous (fuel channels/assemblies, rods, ...)

Fuel Depletion

- Transport theory calculation in unit cell of CANDU 600 (WIMS): depletion at constant power in bundle, at nominal local conditions:
    - exact (hot) geometry of bundle
    - initial fuel composition (natural uranium)
    - fuel density + temperature
    - coolant isotopics + density + temperature
    - moderator isotopics + density + temperature



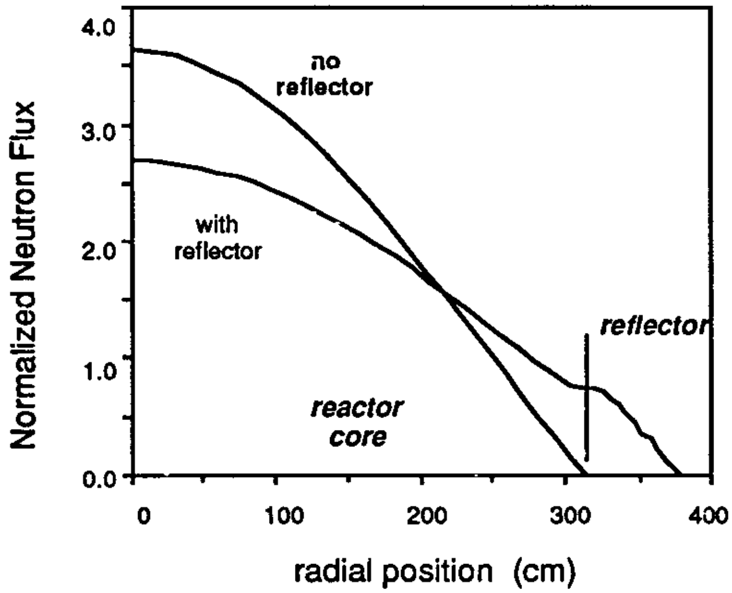
- Fuel burnup distribution is not uniform because neutron flux distribution is non-uniform.
- Significant change in cell multiplication factor (k_∞) with burnup.
- Fuel burnup distribution in reactor will affect the flux shape (i.e. problem is nonlinear)

Flux Flattening

- Typical safety limits CANDU 600:
  - 8.3 MW (chanel)
  - 950 kW (bundle)

- homogeneous core:

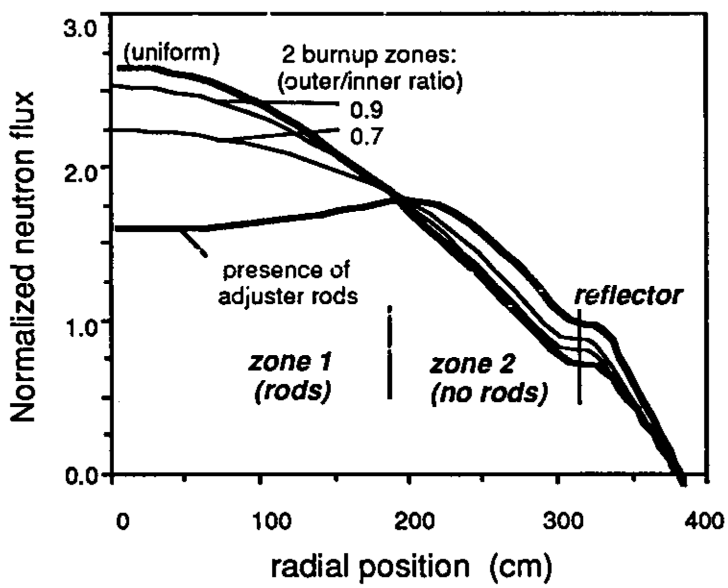
$$\phi(r,z) = \phi_o J_o\left(\frac{2.405}{R}r\right)\cos\left(\frac{\pi}{H}z\right)$$



presence of reflector

12.6 MW ⇒ 9.4 MW
(channel)

- heterogeneous core (2 zones); flattening due to:
  - burnup difference (0.7): 9.4 MW ⇒ 8.7 MW
  - adjuster rods: 8.7 MW ⇒ 6.4 MW



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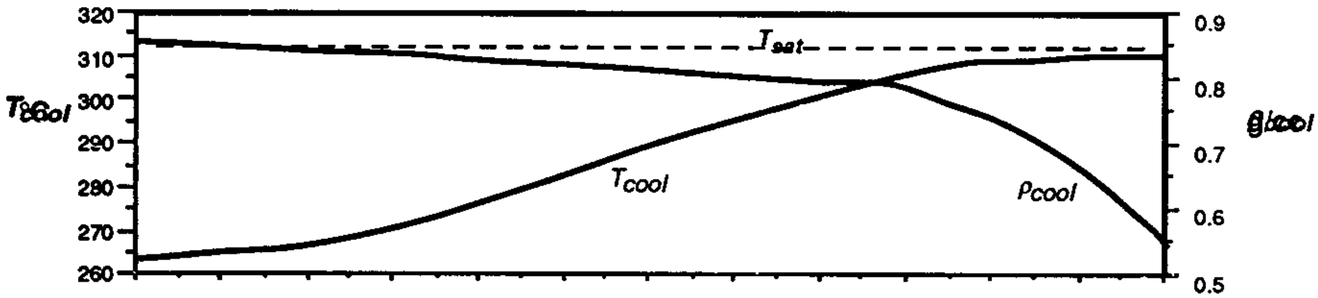
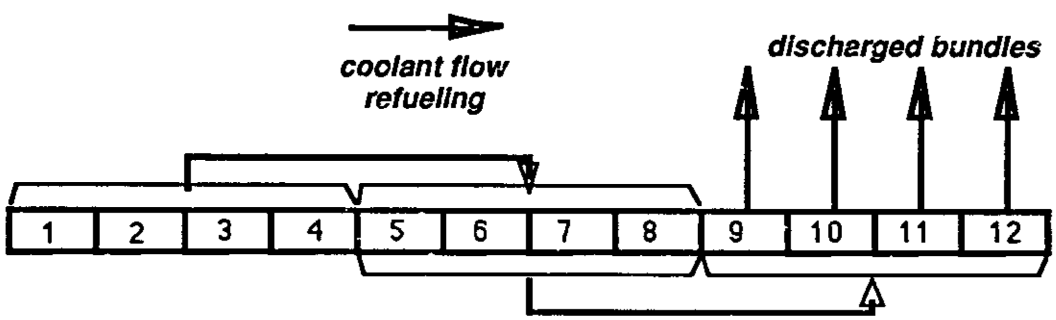
On-Power Refueling

- simple push-through axial shuffling scheme
  - Ex.: 4-bundle scheme (4BS), refueling with flow:





- local parameters of unit cells (bundles) change upon refueling, introducing a history effect in fuel depletion
  - distributed parameters along channel:
    - ¤ fuel temperature
    - ¤ coolant density
    - ¤ coolant temperature
  - axial distribution of neutron flux along channel is not known in unit cell (It is obtained from a seperate diffusion theory calculation for the whole reactor)

Fuel Management Effects

- flow and refueling are *bi-directional* , i.e. alternating direction in neighbouring channels

- *fixed* axial shuffling scheme (uniform or in 2 radial zones)

- channel refueling frequency is *arbitrary*, but average fueling rate must compensate the reactivity decline due to fuel burnup in reactor, within the limits of the fuel handling system.

- a specific fuel management strategy has a characteristic effect on neutron flux and fission power distributions:

  - **zone control level response to refuelings**

  - **Channel Power Peaking Factor (CPPF),** fuel management «ripple»

- an equilibrium refueling model (Time-Average) can be defined to predict fuel management effects

Equilibrium Refueling Model

- When a *fixed fuel management strategy* is applied over time, the refueling frequency eventually (6 months) becomes *constant* whithin regions of the core. A *refueling cycle* (period T_j) is thus defined in *each* channel j

- by definition, instantaneous channel power and bundle power vary around the *time-average* values: P_j and p_{jk}.

- at equilibrium refueling, local (bundle k) macroscopic cross sections are averaged over the refueling cycle (in channel j):

$$\Sigma_{jk} = \frac{1}{\Delta B_{jk}} \int_{B_{jk}^{(BOC)}}^{B_{jk}^{(EOC)}} \Sigma(\theta)\, d\theta$$

  - the <u>bundle burnup increment</u> during the cycle is simply:

$$\begin{aligned} \Delta B_{jk} &= B_{jk}^{(EOC)} - B_{jk}^{(BOC)} \\ &= p_{jk} \cdot T_j \end{aligned}$$

  - the <u>energy produced in channel</u> j during the cycle:

$$\begin{aligned} E_j &= P_j \cdot T_j \\ &= n_j \cdot B_j \end{aligned}$$

    where B_j is the average exit burnup (in MWd/TeU) and n_j is the number of bundles removed at each refueling (axial bundle shift);

  - therefore:

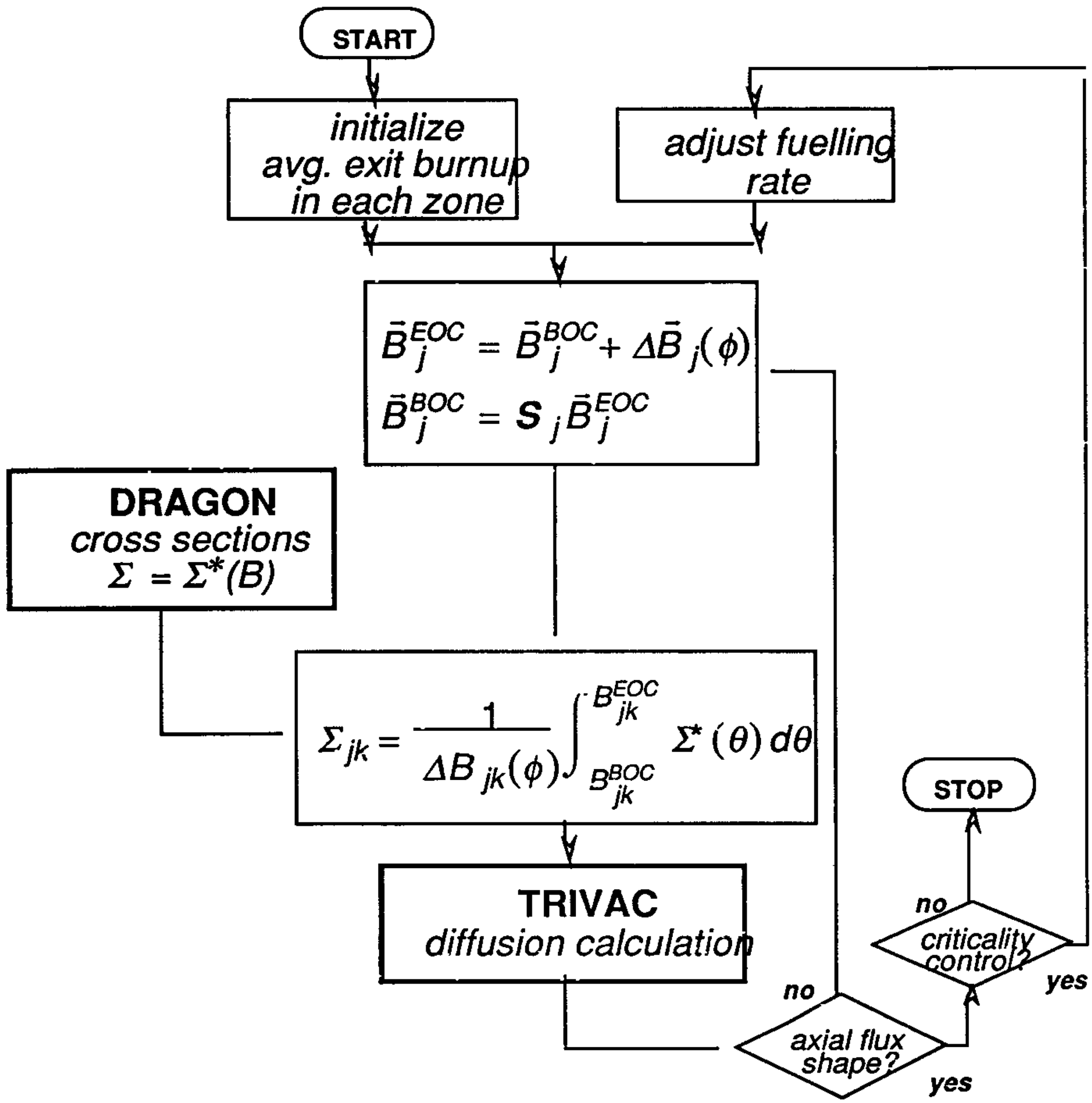
$$\begin{aligned} \Delta B_{jk} &= B_{jk}^{(EOC)} - B_{jk}^{(BOC)} \\ &= n_j B_j \psi_{jk} \end{aligned}$$

    where ψ_{jk} is the *axial power (flux) shape*

$$\psi_{jk}(\phi) = \frac{p_{jk}}{P_j} = \frac{\text{bundle power}}{\text{channel power}}$$

- **the T/A procedure is *nonlinear***

Time-Average Procedure in Reactor Calculation



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Instantaneous Power Distribution

- *power limits* on fuel must never be exceeded (instantaneous power distribution)
- due to fuel depletion, refuelings and controller movements, actual *channel and bundle power varies with time*;
- *peak channel and bundle powers* are necessarily *higher* than the maximum T/A values;
- any realistic instantaneous power distribution shows *a «fuel management ripple»*:
  - function of the refueling perturbation
    - *bundle shift* (fraction of channel refueled)
    - *fresh fuel enrichment* (fissile isotopes)
    - *average exit burnup* (fuel depletion)
- channel «age» model:
  - an instantaneous value of burnup can be assigned to each bundle:

$$B_{jk}(t) = B_{jk}^{BOC} + f_j(t) \cdot \Delta B_{jk}\left(\phi_{T/A}\right)$$

    where $f_j(t)$ is the «age» of channel j $(0 < f_j < 1)$
  - the distribution of $f_j(t)$ must reflect a *refueling sequence*
  - diffusion calculation with $\Sigma(B_{jk})$ yields an *instantaneous* power distribution, $P_j(t)$

Channel Power Peaking Factor (CPPF)

- <u>definition</u>:

  maximum ratio of instantaneous to T/A channel power

$$CPPF(t) = \max_j \left\{ \frac{P_j(t)}{\bar{P}_j} \right\}$$

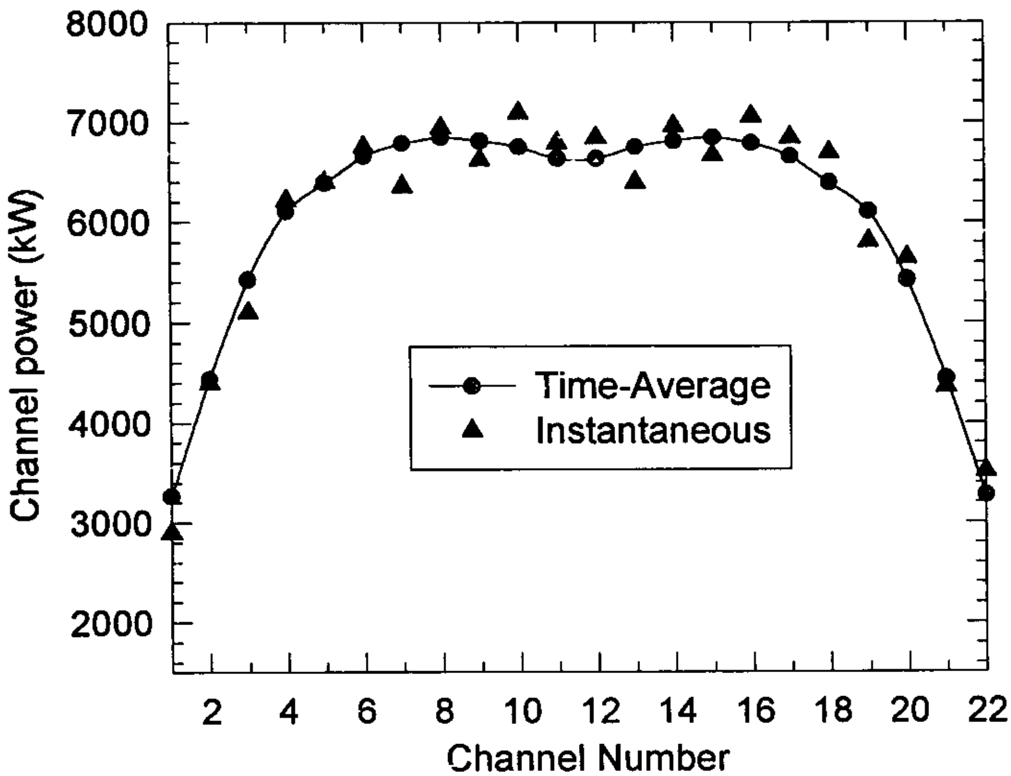
- for current CANDU's, CPPF < 1.15

- varies according to actual *refueling history* in operating reactor
  - axial bundle shift
  - initial enrichment
  - exit burnup
  - refueling sequence

- must be evaluated for ROPT recalibration during operation (for continued protection against dryout)

- must be evaluated during design (to verify safety margins)

Instantaneous vs Time-Average Power Distributions



Radial Channel Power Profiles
(NAT. U, 8BS, Row L)

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2- Reactor Physics Analysis of CANDU Reactors

- A complete suite of reactor physics codes must be used to analyse the fuel management behaviour of a reactor, dealing successfuly with:

  - heterogeneity of materials
  - variety and complexity of neutron cross sections
  - neutron transport and streaming
  - temperature feedback
  - radioactive decay...

- This is acheived through a breakdown of the task at two levels, carrying out seperate lattice and reactor calculations for the neutron field:

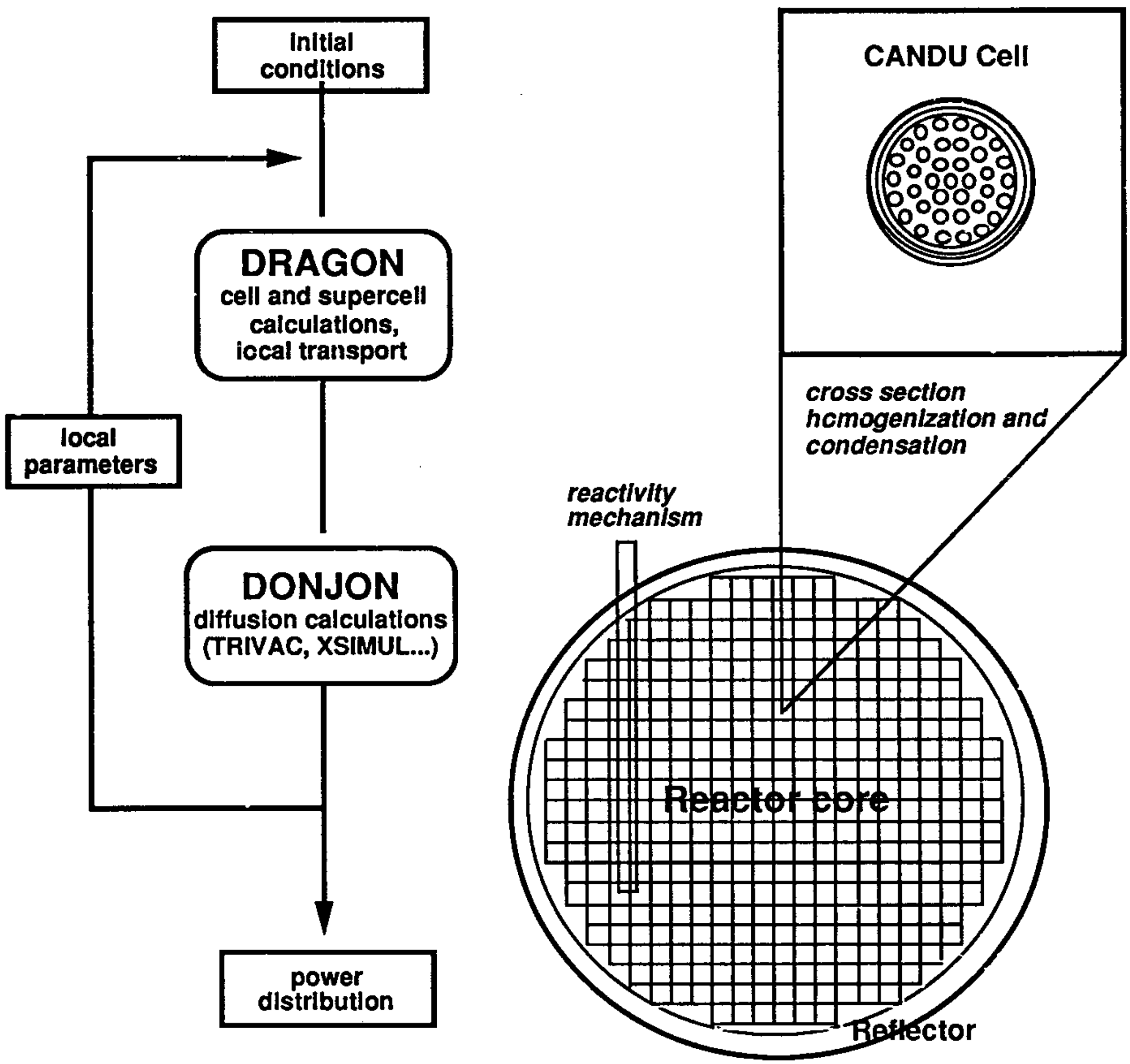
  1. Using an appropriate microscopic cross section library, the transport code DRAGON solves the integral transport theory equation over an infinite lattice of unit cells (in 2D) and various supercells (in 3D), using many neutron energy groups;

  2. Knowledge of the detailled flux within these cells allows DRAGON to homogenize the macroscopic cross sections over the cells (bundles), and to condense the nuclear properties down to a few energy groups (usually 2);

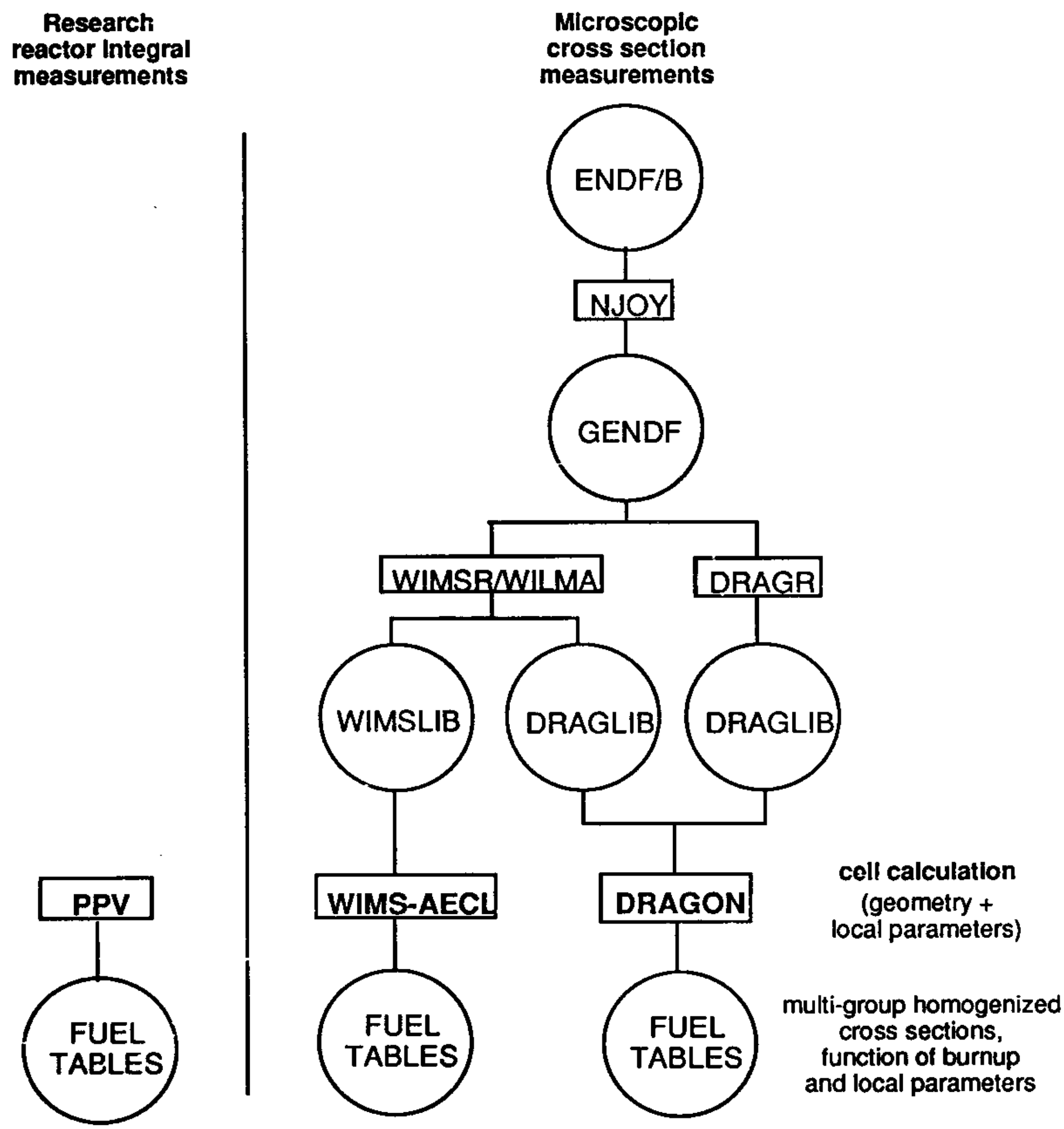
  3. Reactor code DONJON can access the data tabulated by DRAGON to construct a diffusion theory model of the entire reactor, including fuel shuffling and control rod motion. The 3D multigroup diffusion module TRIVAC can solve for direct, adjoint, generalized adjoint and harmonic flux distributions in cartesian and hexagonal geometries.

  4. Fuel management needs to predict fuel burnup accurately. Because of the separation in the nuclide field, separate cell depletion and global depletion calculations are required, with an inherent difficulty in dealing with the distributed local parameters.

Separate Lattice and Core Calculations

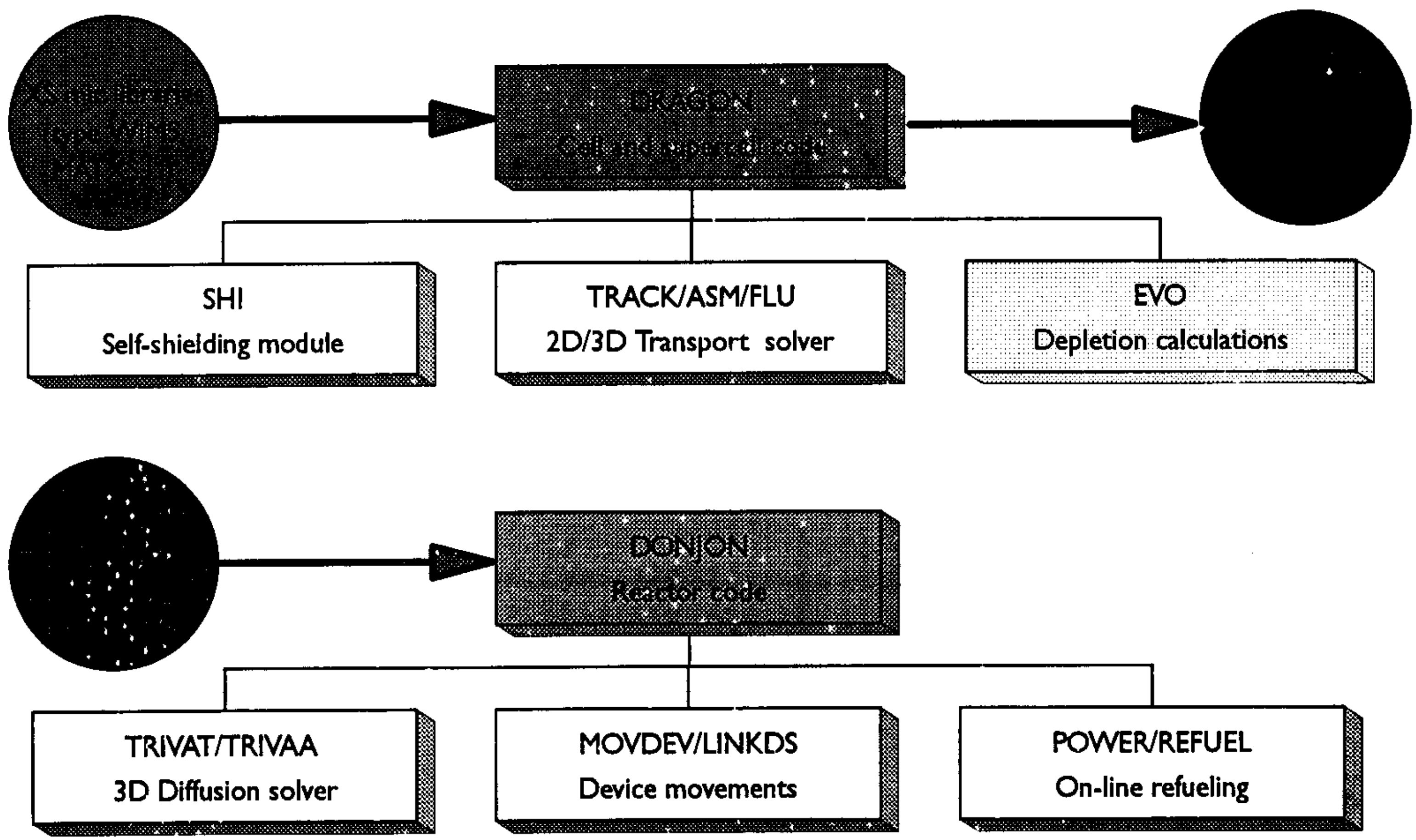


Chaining of Cross Section Processing Codes

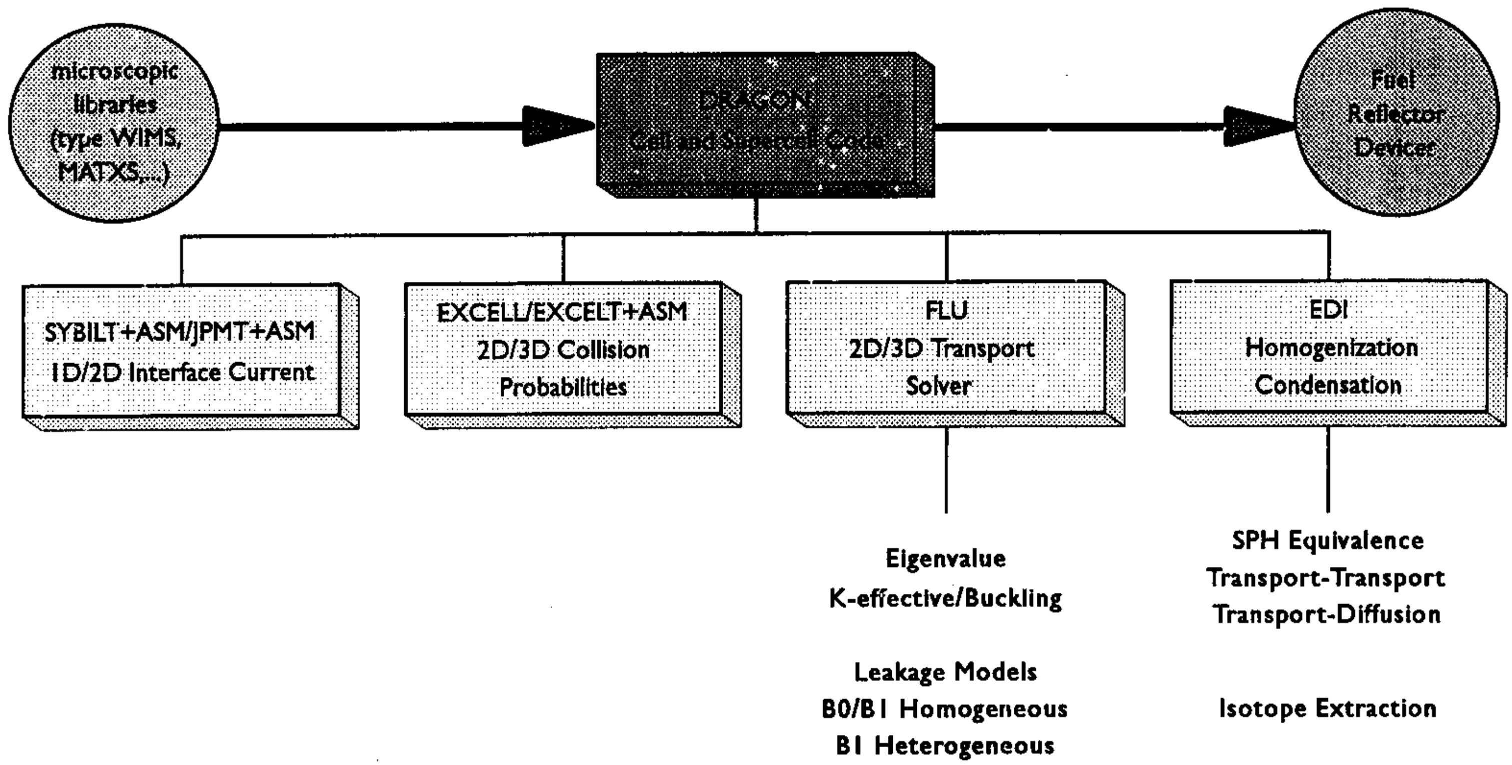


Reactor Physics Chain of Codes at Polytechnique

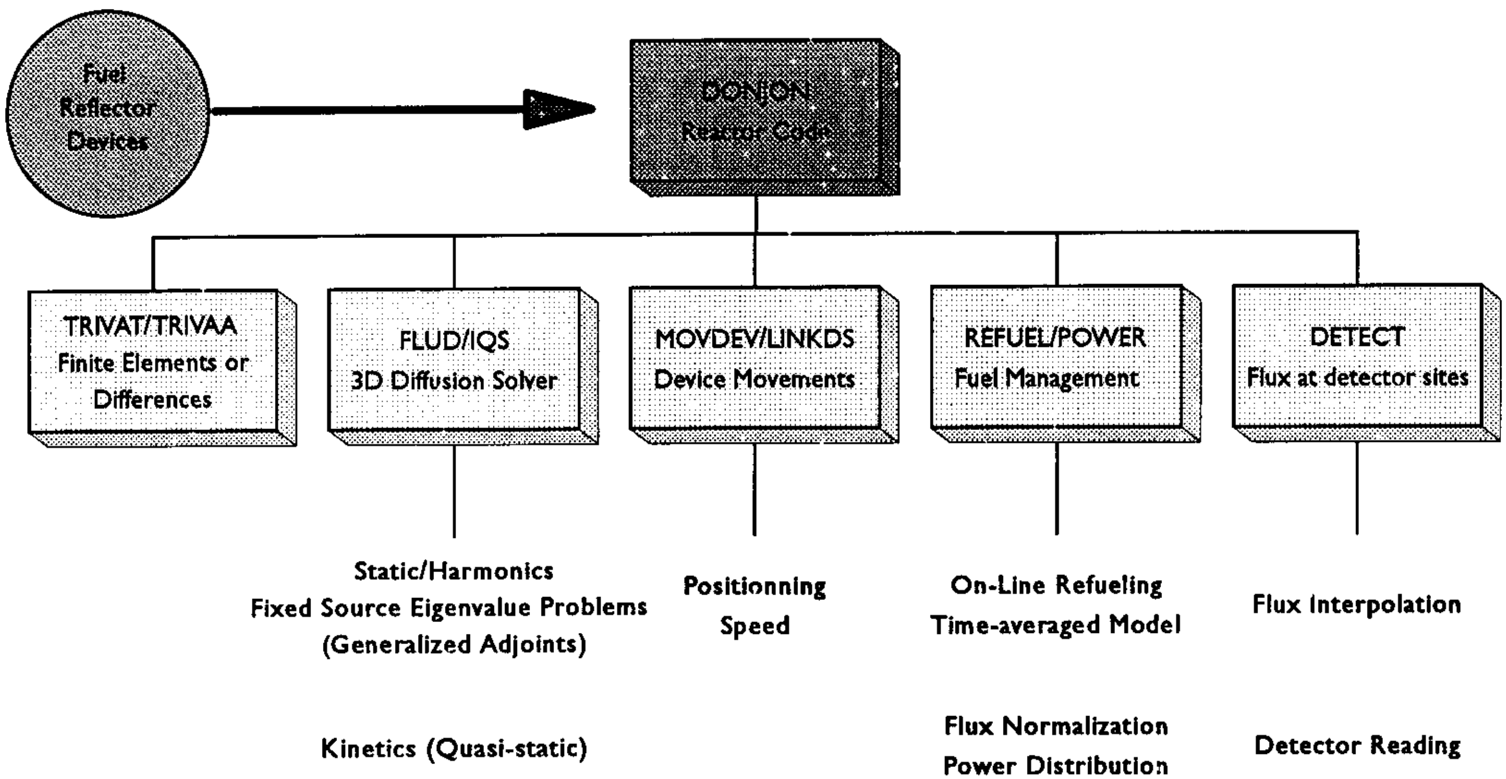
CLE-2000 language, GAN driver



DRAGON Transport Code

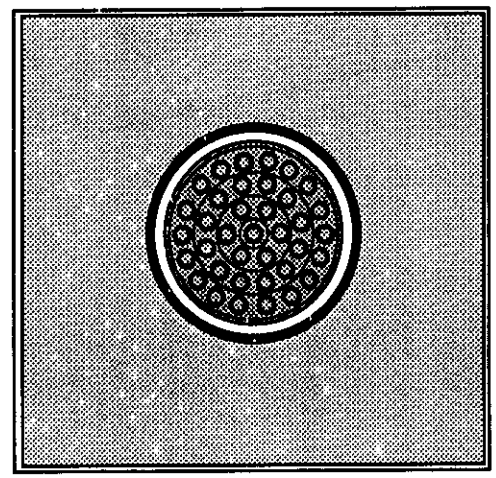


DONJON Diffusion Code



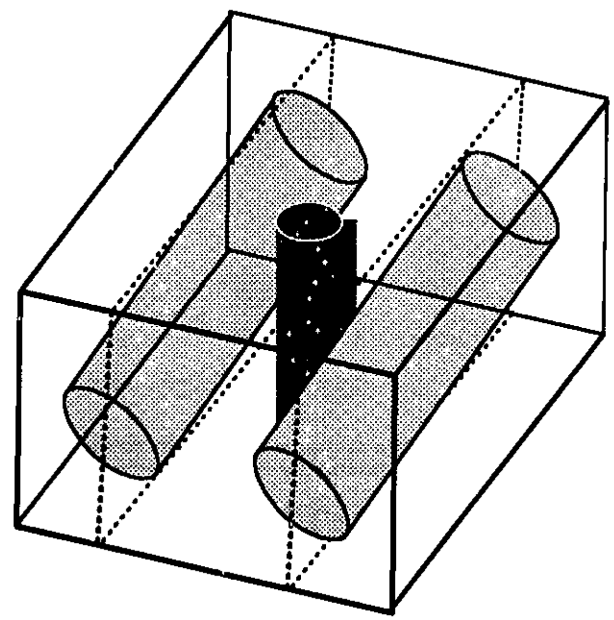
DRAGON Macroscopic Cross Sections

- CANDU Cell Calculations (2D)
  - Bundle Local Parameters
  - Self-Shielding
  - Flux Calculation with Critical Buckling Search
  - Depletion Calculations
  - Condensation/Homogenization



DRAGON Macroscopic Cross Sections (next)

- Supercell Calculations (3D)
  - Multigroup Properties
  - 3D Geometry for Devices
  - Flux Calculation with Critical Buckling Search
  - Condensation/homogenization



3. Generalized Perturbation Theory (GPT)

- GPT is widely used in LWR Fuel Management Analysis
  - first and second order estimates of system characteristics (arbitrary flux functionals)
  - optimization of reload patterns
  - usually 2D analysis

- Use of GPT in CANDU
  - developped at École Polytechnique
  - 1992: application of first order GPT in 3D to simultaneous optimization of adjuster layout and equilibrium refuelling in OPTEX-4 (core design)
  - 1997: prediction of zone controller level response to refuelling perturbations in DONJON (selection of refuelling sites)

- Future Application
  - advanced fuel cycles evaluation, uncertainty due to heterogeneity effect in DUPIC, optimization of refuelling sequence.

Generalized Adjoints

- Examples of flux functionals

  - zonal power (ratio) in zone ℓ

$$p_\ell(\vec{X},\phi) = \frac{\langle H,\phi\rangle_{V_\ell}}{\langle H,\phi\rangle_R}$$

fuelling costs

$$F(\vec{X},\phi) = \frac{\left\langle \frac{CH}{B},\phi \right\rangle}{\langle H,\phi\rangle}$$

where $\vec{X}$ is a state vector (enrichment, exit burnup, poisons, ...). H, C, B are functions of $\vec{X}$.

- Unperturbed Flux Distribution (diffusion equation)

$$\boxed{\boldsymbol{M}\,\phi_o \;=\; \lambda\,\boldsymbol{F}\,\phi_o}$$

- Perturbations in channel (or zone) j

  - if ***ΔM(j)*** and ***ΔF(j)*** are changes in diffusion equation operators due to a perturbation in region j (ex. refuelling), the zone power response in zone ℓ is:

$$\Delta p_\ell^{(j)} = \Delta p_\ell^{(j)}\Big|_{\Delta\Sigma} + \Delta p_\ell^{(j)}\Big|_{\Delta\phi}$$

where

$$\Delta p_\ell^{(j)}\Big|_{\Delta\Sigma} = \frac{\left\langle \Delta H^{(j)},\phi_o \right\rangle_{V_\ell}}{\langle H,\phi_o\rangle_R} - p_\ell \frac{\left\langle \Delta H^{(j)},\phi_o \right\rangle_R}{\langle H,\phi_o\rangle_R}$$

and

$$\Delta p_\ell^{(j)}\Big|_{\Delta\phi} = -\left\langle \Gamma_\ell^*, \left(\Delta \boldsymbol{M}^{(j)} - \lambda_o \Delta \boldsymbol{F}^{(j)}\right)\phi_o \right\rangle$$

Generalized Adjoint Equation

- Generalized Adjoint is solution to the fixed source eigenvalue problem:

$$\left(\boldsymbol{M}_o^* - \lambda_o \boldsymbol{F}_o^*\right)\Gamma_\ell^* = S_\ell^*$$

where

$$S_\ell^* = \frac{\partial p_\ell}{\partial \phi}(\bar{r}) = \frac{\left[\delta_{V_\ell} - p_\ell\right] H_o(\bar{r})}{\langle H, \phi_o \rangle_R}$$

- Orthogonality condition:

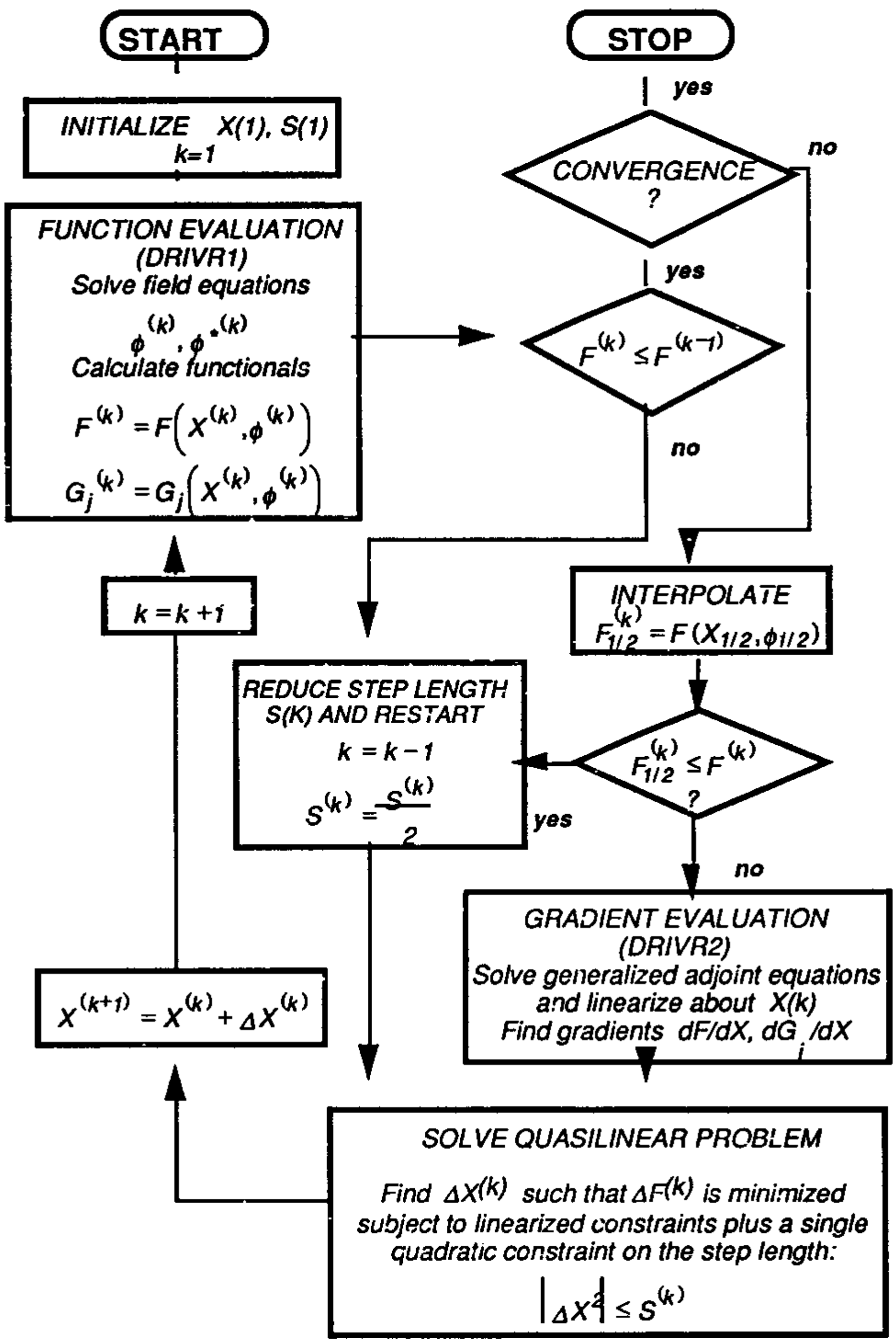
$$\left\langle \Gamma_\ell^*, F_o \phi_o \right\rangle_R = 0$$

- Arbitrary variations of flux functionals can be obtained using unperturbed flux only.

  - application with mathematical programming to solve reactor design optimization problem (OPTEX-4)

  - response of Liquid Zone Controllers to refuelling perturbations

NLP Algorithm in OPTEX-4



4. Global Depletion Schemes

- Seperate lattice calculation creates a problem for treatment of local parameters in core calculation

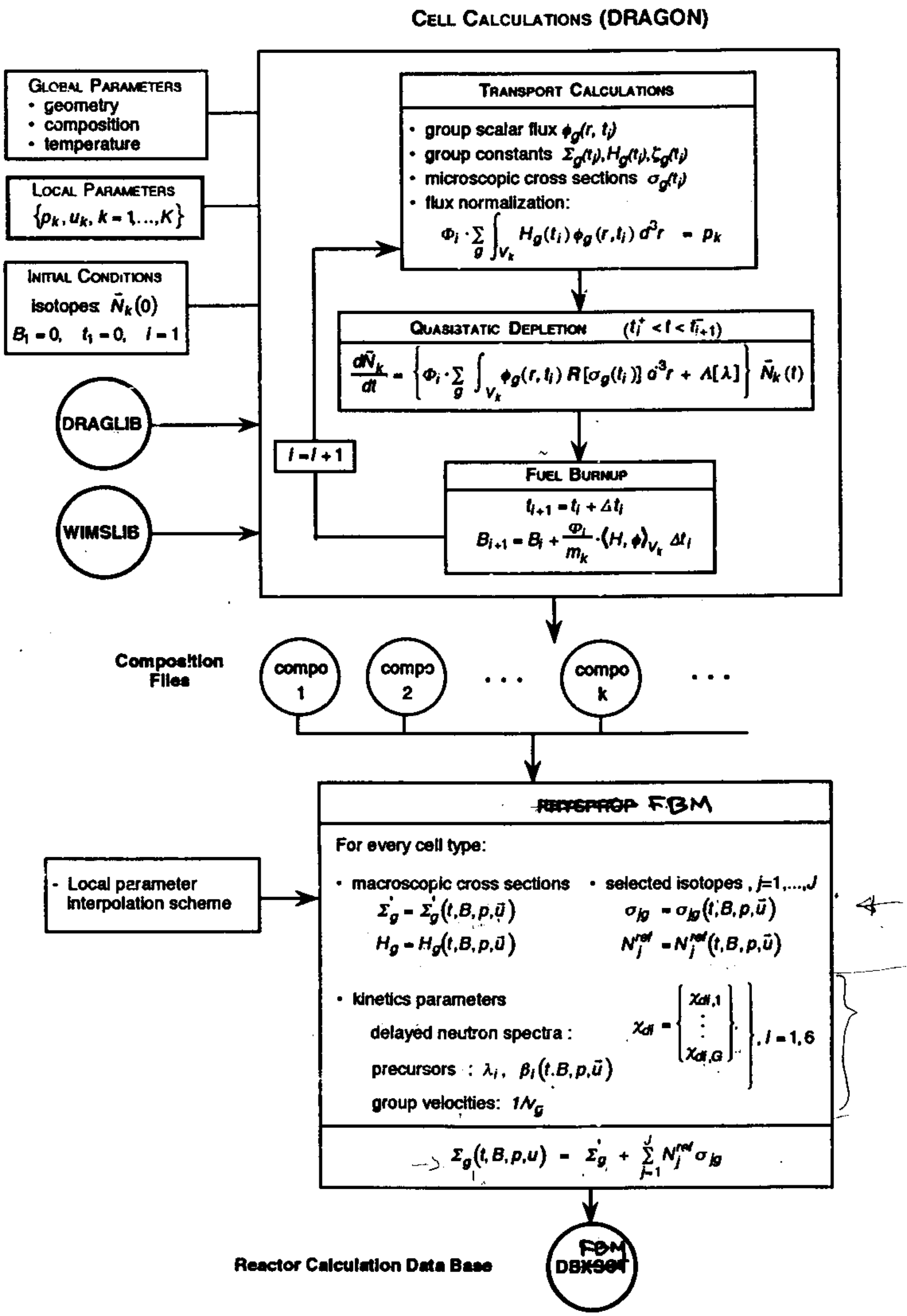
  - introduction of a Feedback Model (FBM) in DRAGON/DONJON

- Block depletion

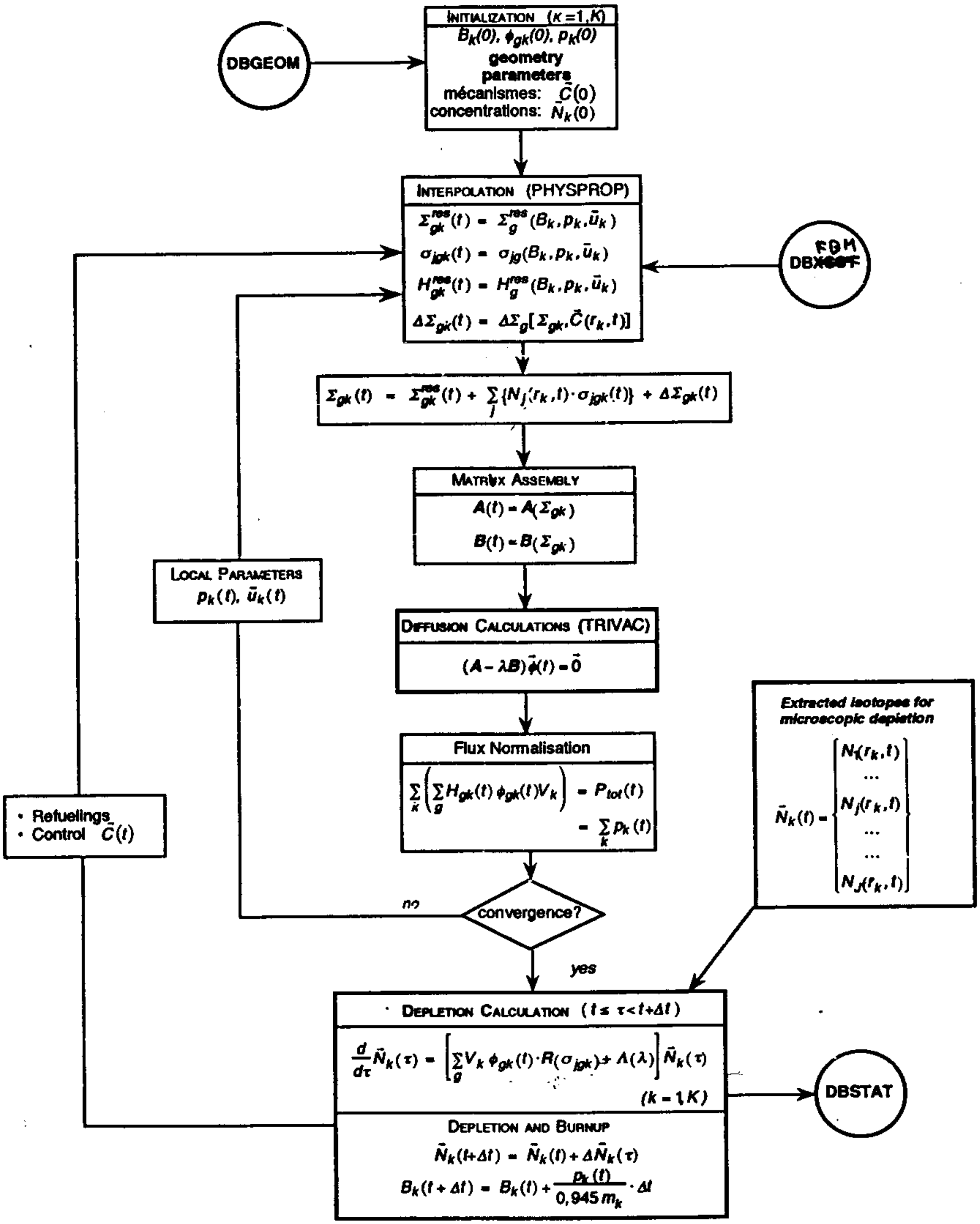
  - large number of unknowns for the nuclide field in global block depletion methods;

  - introduction of a simplified global depletion method

Depletion Calculations in DRAGON



Global Depletion Calculations in DONJON



TOWARDS A GLOBAL DEPLETION CALCULATION METHOD IN DRAGON AND DONJON

G. Marleau, M. Boubcher and H. Benjaafar

1. The Cell Depletion Model of DRAGON
2. A Viable Depletion Model in DONJON
3. A Simplified Reactor Depletion Model

The Cell Depletion Model of DRAGON (1)

- Requirements

  1. A multigroup cross section library

     The multigroup microscopic cross section for each isotope and each reaction type x, $\sigma^g_{x,i}$

  2. A depletion chain coherent with this library

     The decay rate for each isotope λ_i

     The production yield for each isotope $Y_{x,i\rightarrow j}$ for production reaction p including decay ($p = d$)

  3. Global cell parameters

     – Geometry including material composition and mesh decomposition

     – Local parameters including temperature, power, etc.

     – Initial conditions $N_{i,r}(t_o)$ for isotope i in region r at $t = t_o$

The Cell Depletion Model of DRAGON (2)

- Calculation process

1. Multigroup macroscopic material cross section calculation

   Using $N_{i,r}(t)$ at $t = t_o$ and $\sigma^g_{x,i}$

   $$\Sigma^g_{x,r}(t) = \sum_i N_{i,r}(t)\sigma^g_{x,i}$$

2. Flux calculation

   Using $\Sigma^g_{x,r}(t)$, solve the transport equation using collision probability method for flux ϕ^g_r in each region r for each energy group g.

3. Depletion calculation

   Solve the isotopic depletion for $N_{j,r}(t_f)$ giving the final concentration of each nuclide j in region r at $t = t_f$ assuming a constant power of flux between time t_o and t_f, namely:

   $$\begin{aligned}\frac{dN_{j,r}(t)}{dt} &= \sum_i N_{i,r}(t) \sum_x Y_{x,i\to j} \sum_g \phi^g_r \sigma^g_{x,i} \\ &+ \sum_i N_{i,r}(t) Y_{d,i\to j} \lambda_i\end{aligned}$$

4. Go back to step 1

The Cell Depletion Model of DRAGON (3)

- Dimension of the cell problem:

  1. Total number of regions:

     Relatively small (31 for a typical CANDU cell)

  2. Number of fuel regions:

     Relatively small (4 for a CANDU cell)

  3. Number of groups:

     Relatively large (69 or 89 groups for our libraries)

  4. Number of isotopes per region:

     Relatively large (from 50 to 100 depending on the library)

A Viable Depletion Model in DONJON

- Dimension of the reactor problem with fuel depletion model

  1. Total number of regions:

     Very large (at least 5×4560 regions for a CANDU with 4 fuel regions and a moderator region)

  2. Number of groups:

     Relatively small (generally 2 groups for CANDU)

  3. Number of isotopes per region:

     Relatively large (from 50 to 100 depending on the library)

- Total number of unknowns in depletion equation $N_d = N_f N_c N_i$

  Number of cells: $N_c = 4560$

  Number of fuel regions per cell: $N_f = 4$

  Number of isotopes per fuel region: $N_i = 50$

  Total number of unknowns $N_d = 912000$

A Simplified Reactor Depletion Model (1)

- Combine the N_f fuel regions into a single region

$$F_i^g(t) = \frac{\sum_r N_{i,r}(t)\phi_r^g(t)}{N_i(t)\phi^g(t)}$$

with $\phi^g(t)$ the average flux in the cell and $N_i(t)$ the average concentration of isotope i in the cell.

- New depletion equation:

$$\begin{aligned}\frac{dN_j(t)}{dt} &= \sum_i N_i(t)\sum_x Y_{x,i\to j}\sum_g F_i^g(t)\phi^g\sigma_{x,i}^g \\ &+ \sum_i N_i(t)Y_{d,i\to j}\lambda_i\end{aligned}$$

A Simplified Reactor Depletion Model (2)

- Combine many fission products into a pseudo-fission product

$$
\begin{aligned}
N_{\mathrm{pf}}(t) &= \sum_{i\in \mathrm{pf}} N_i(t) \\
Y_{x,i\to \mathrm{pf}} &= \sum_{j\in \mathrm{pf}} Y_{x,i\to j} \\
Y_{x,\mathrm{pf}\to \mathrm{pf}} &= \frac{\sum_{i\in \mathrm{pf}} \sum_{j\in \mathrm{pf}} N_i(t) Y_{x,i\to j}}{N_{\mathrm{pf}}(t)} \\
\lambda_{\mathrm{pf}} &= \frac{\sum_{i\in \mathrm{pf}} \sum_{j\in \mathrm{pf}} N_i(t) Y_{d,i\to j}\lambda_i}{N_{\mathrm{pf}}(t) Y_{d,\mathrm{pf}\to \mathrm{pf}}}
\end{aligned}
$$

- New depletion equation for pseudo-fission product:

$$
\begin{aligned}
\frac{dN_{\mathrm{pf}}(t)}{dt} &= \sum_i N_i(t) \sum_x Y_{x,i\to \mathrm{pf}} \sum_g F^g(t)\phi^g \sigma^g_{x,i} \\
&+ \sum_i N_i(t) Y_{d,i\to \mathrm{pf}} \lambda_i \\
&+ N_{\mathrm{pf}}(t) \sum_x Y_{x,\mathrm{pf}\to \mathrm{pf}} \sum_g F^g(t)\phi^g \sigma^g_{x,\mathrm{pf}} \\
&+ N_{\mathrm{pf}}(t) Y_{d,\mathrm{pf}\to \mathrm{pf}} \lambda_{\mathrm{pf}}
\end{aligned}
$$

A Simplified Reactor Depletion Model (3)

- Combine many heavy isotopes (actinides) into a few pseudo-actinides

$$
\begin{aligned}
N_{\mathsf{pa}}(t) &= \sum_{i \in \mathsf{pa}} N_i(t) \\
Y_{x,i \to \mathsf{pa}} &= \sum_{j \in \mathsf{pa}} Y_{x,i \to j} \\
Y_{x,\mathsf{pa} \to \mathsf{pa}} &= \frac{\sum_{i \in \mathsf{pa}} \sum_{j \in \mathsf{pa}} N_i(t) Y_{x,i \to j}}{N_{\mathsf{pa}}(t)} \\
\lambda_{\mathsf{pa}} &= \frac{\sum_{i \in \mathsf{pa}} \sum_{j \in \mathsf{pa}} N_i(t) Y_{d,i \to j} \lambda_i}{N_{\mathsf{pa}}(t) Y_{d,\mathsf{pa} \to \mathsf{pa}}}
\end{aligned}
$$

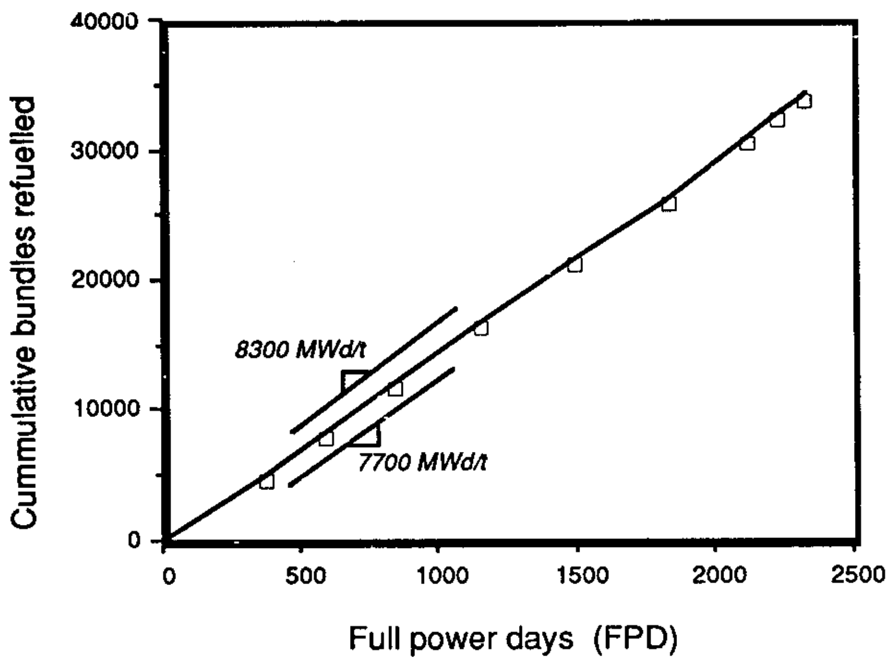
- New depletion equation for pseudo-actinide:

$$
\begin{aligned}
\frac{dN_{\mathsf{pa}}(t)}{dt} &= N_{\mathsf{pa}}(t) \sum_x Y_{x,\mathsf{pa} \to \mathsf{pa}} \sum_g F^g(t) \phi^g \sigma^g_{x,\mathsf{pa}} \\
&+ N_{\mathsf{pa}}(t) Y_{d,\mathsf{pa} \to \mathsf{pa}} \lambda_{\mathsf{pa}}
\end{aligned}
$$

5. Advanced Fuel Cycles in CANDU

- use of natural uranium:

  - low fuel costs
    - simple fuel cycle (once-through)
    - reliable fuel handling and safe storage
  - daily refueling requires very small excess reactivity
    - no soluble poison (boron) in moderator
    - average exit burnup nearly 2x average burnup (continuous refueling limit)
  - average exit burnup acheived: 7500-8000 MWd/TeU
    - low fissile content of spent fuel (≈2.0 g/kg U-235, 2.4 g/kg Pu-239)
    - no commercial reprocessing



Actual average exit burnup in Gentilly-2

Advanced fuel cycles

- uranium enrichment
  - availability of U enrichment services
  - recovered uranium (RU) from commercial reprocessing of PWR fuel
  - modest fuel fabrication costs
  - improved resource utilization (higher burnup, lower fuel costs)

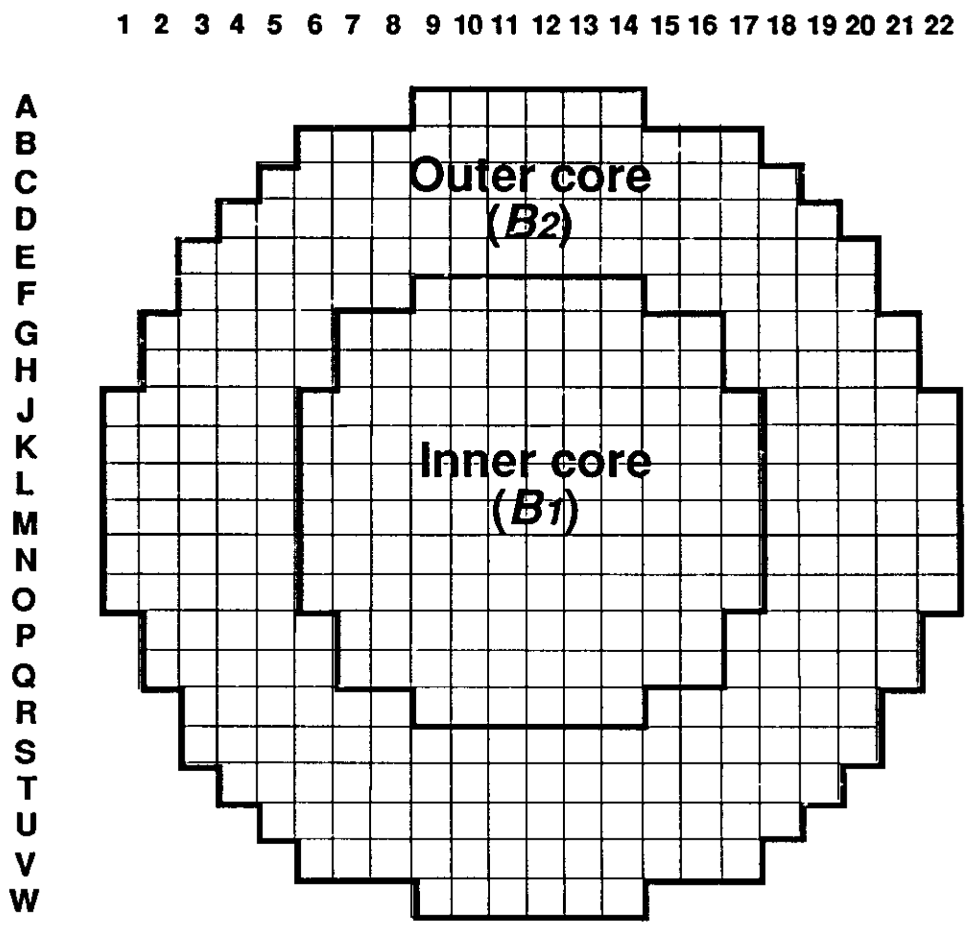
- MOX
  - availability of from military Pu disposal (or commercial reprocessing of PWR fuel)
  - feasability study, reactor physics experiments

- DUPIC
  - direct use of PWR fuel in CANDU
  - dry reprocessing
  - OREOX process
  - reactor physics studies
  - heterogeneity problem

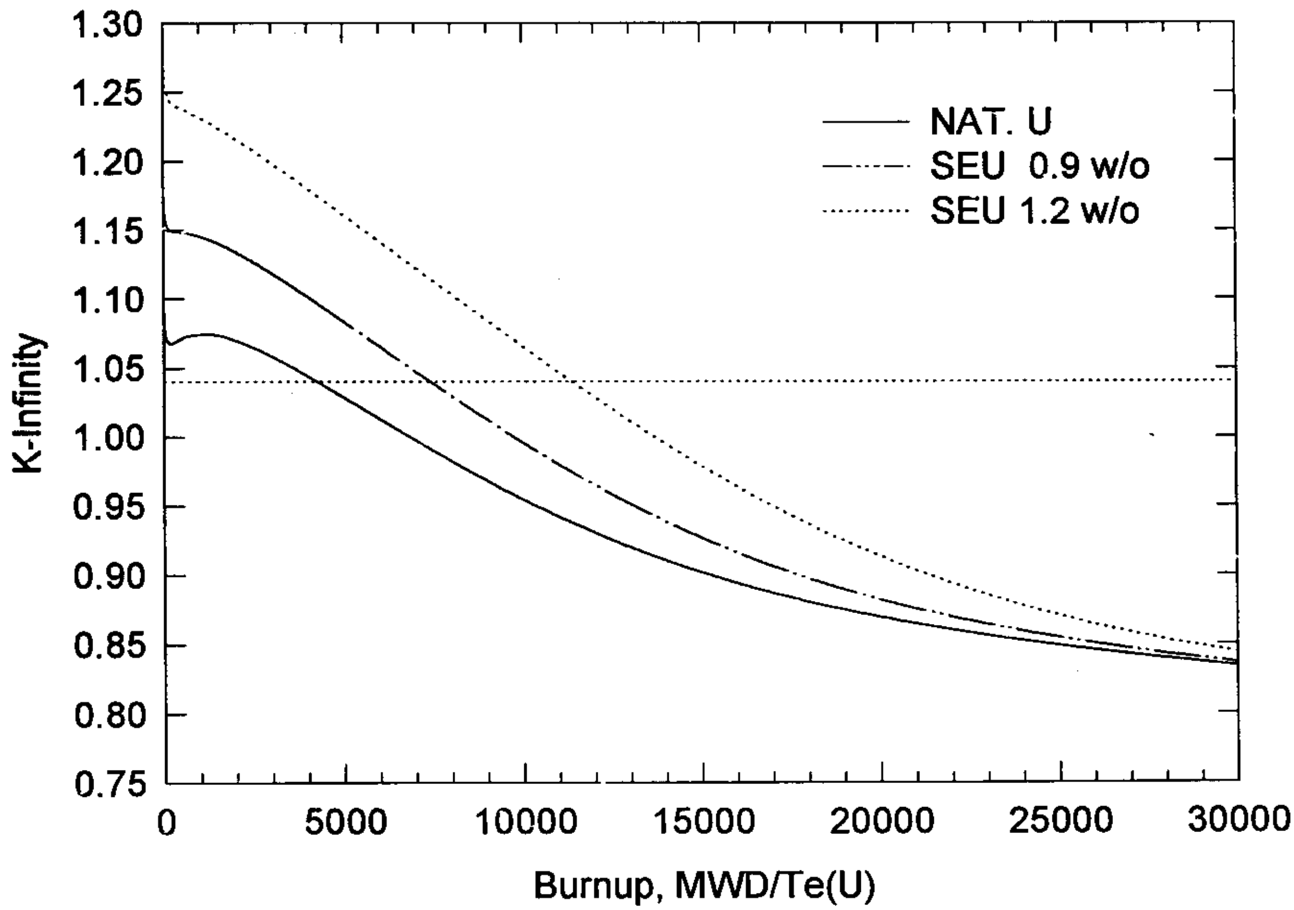
- DRAGON/DONJON results
  - compare various cycles at equilibrium refueling for different axial shuffling schemes
    - average exit burnup
    - channel and bundle power distributions
    - adjuster worth
    - CPPF
  - study heterogeneity problem in DUPIC (carrying out both PWR and CANDU depletion calculations)
  - full core CANDU 6 model, with adjusters and LZC
  - age model

Time Average Calculations

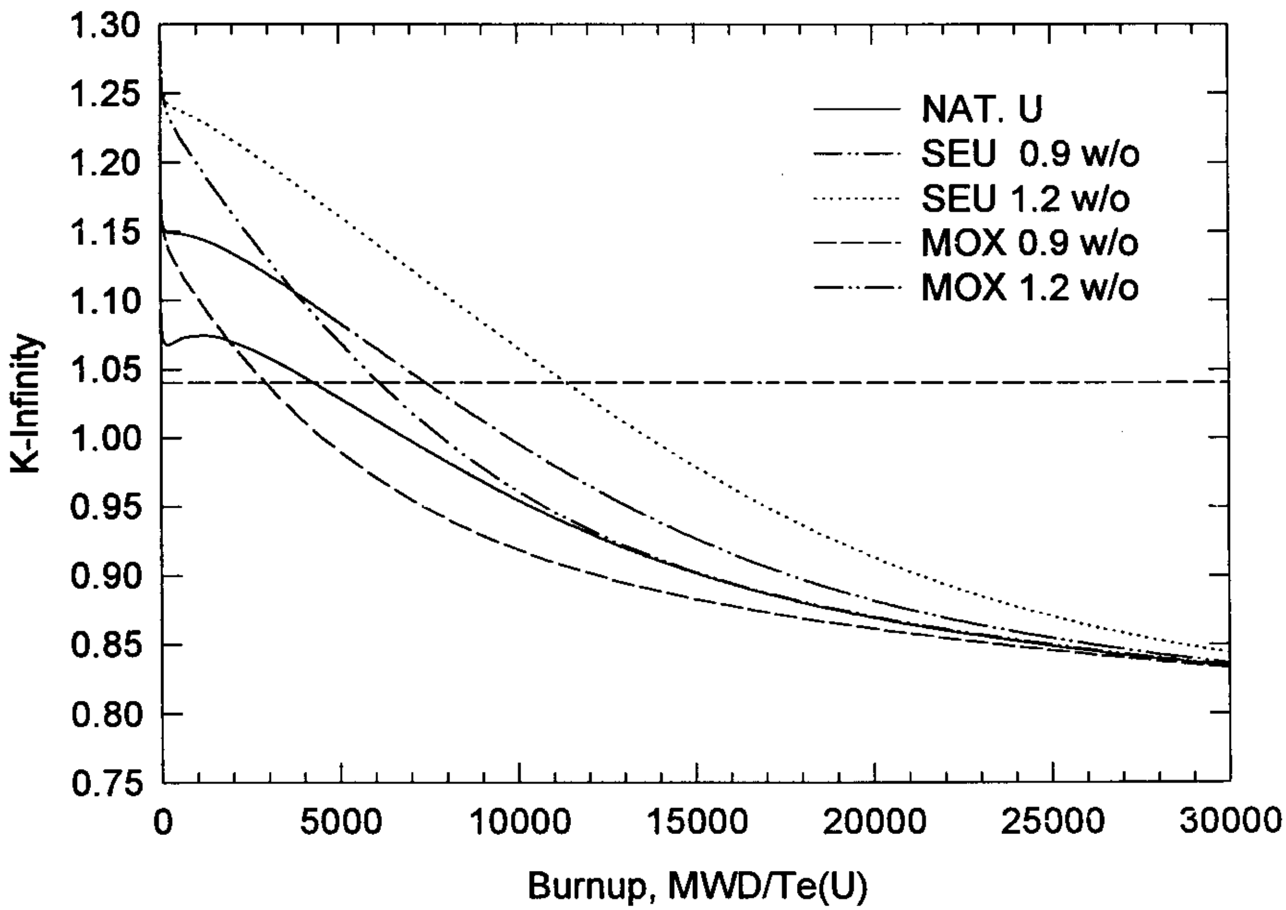
- unit cell depletion calculations with DRAGON:
  - o 37-element fuel (Nat.U, SEU, MOX)
  - o CANFLEX (DUPIC)
- nominal 615 kW bundle, local parameters
- 2-zone differential refueling (inner/outer core)



- *equilibrium refueling*: average exit burnup is adjusted to make T/A model critical
- ratio of inner core/outer core burnup adjusted to minimize peak power (radial flattening)



Lattice K-infinity vs. Burnup for Different Fuels



Lattice K-infinity vs. Burnup for Different Fuels

Numerical Results

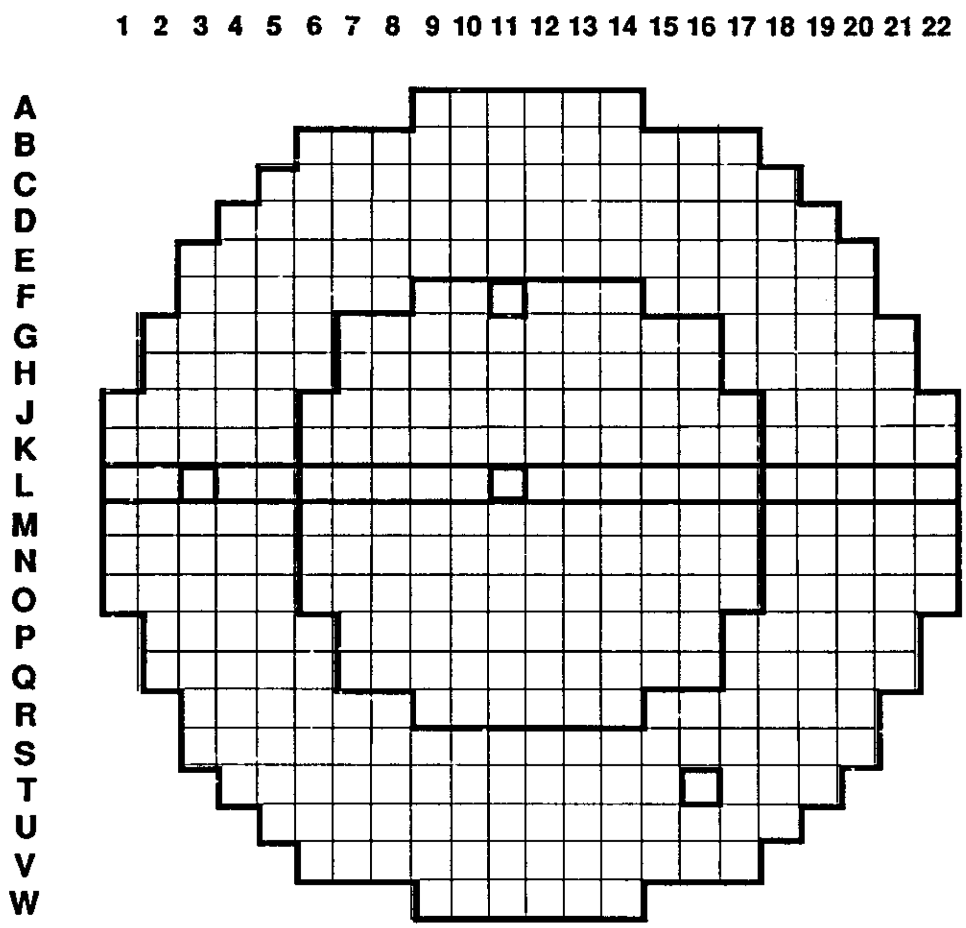
- large increase in initial k_∞ with SEU $\Rightarrow$ higher burnup
- MOX: match initial k_∞ of SEU
  - o Pu is added (uniformily) to depleted uranium (0.2%)
  - o k_∞ drops more rapidely $\Rightarrow$ lower burnup than SEU
- peak channel and bundle power sensitive to axial BS

Characteristics of CANDU-6 Core Refueled by Different Fuel Types

FUEL TYPE	*BS*	*AVERAGE EXIT BURNUP MWd/T(U)*	*PEAK CHANNEL POWER (MW)*	*PEAK BUNDLE POWER (kW)*	*ADJUSTER WORTH (mk)*
Nat. U	2	7710	7252 (J11)	868 (E12,6)	17.49
	4	7668	7231 (J11)	839 (E12,6)	17.62
	8	7613	6991 (H8)	837 (E12,6)	16.64
SEU 0.9 w/o	2	14226	6738 (F15)	777 (E11,7)	14.00
	4	14292	6748 (F15)	799 (E14,5)	14.43
	8	14768	6860 (H15)	856 (E12,6)	15.43
SEU 1.2 w/o	2	21715	6939 (E14)	865 (F15,3)	10.14
	4	22169	6806 (F8)	872 (F8,9)	12.30
	8	23799	6840 (H8)	867 (G8,7)	14.99
MOX 0.9 w/o	2	5931	6806 (E14)	732 (F15,4)	12.47
	4	6051	6773 (F15)	775 (F15,4)	13.44
	8	6327	6750 (G15)	859 (E12,6)	14.57
MOX 1.2 w/o	2	12454	6844 (F15)	850 (F15,3)	10.34
	4	12796	6859 (E14)	854 (F15,4)	10.61
	8	13907	6862 (H15)	865 (E11,7)	14.54

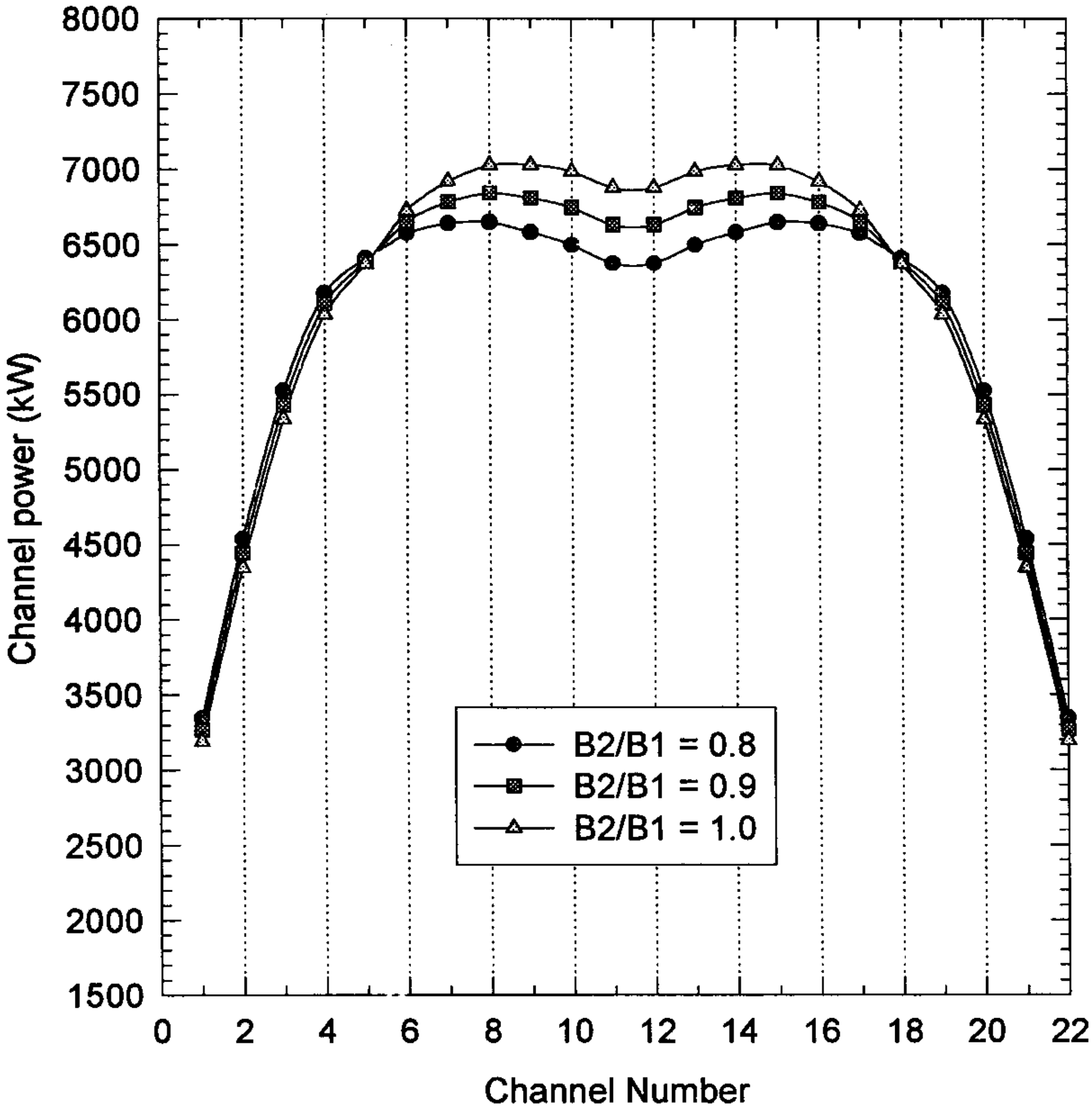
Power Distribution

- size of inner burnup zone not optimized for enriched core
- axial power profiles in selected channels:
  - o axial offset increases with fuel enrichment
  - o axial offset larger as BS is reduced (8BS ⇒ 4BS ⇒ 2BS)
  - o flattenning due to presence of adjusters

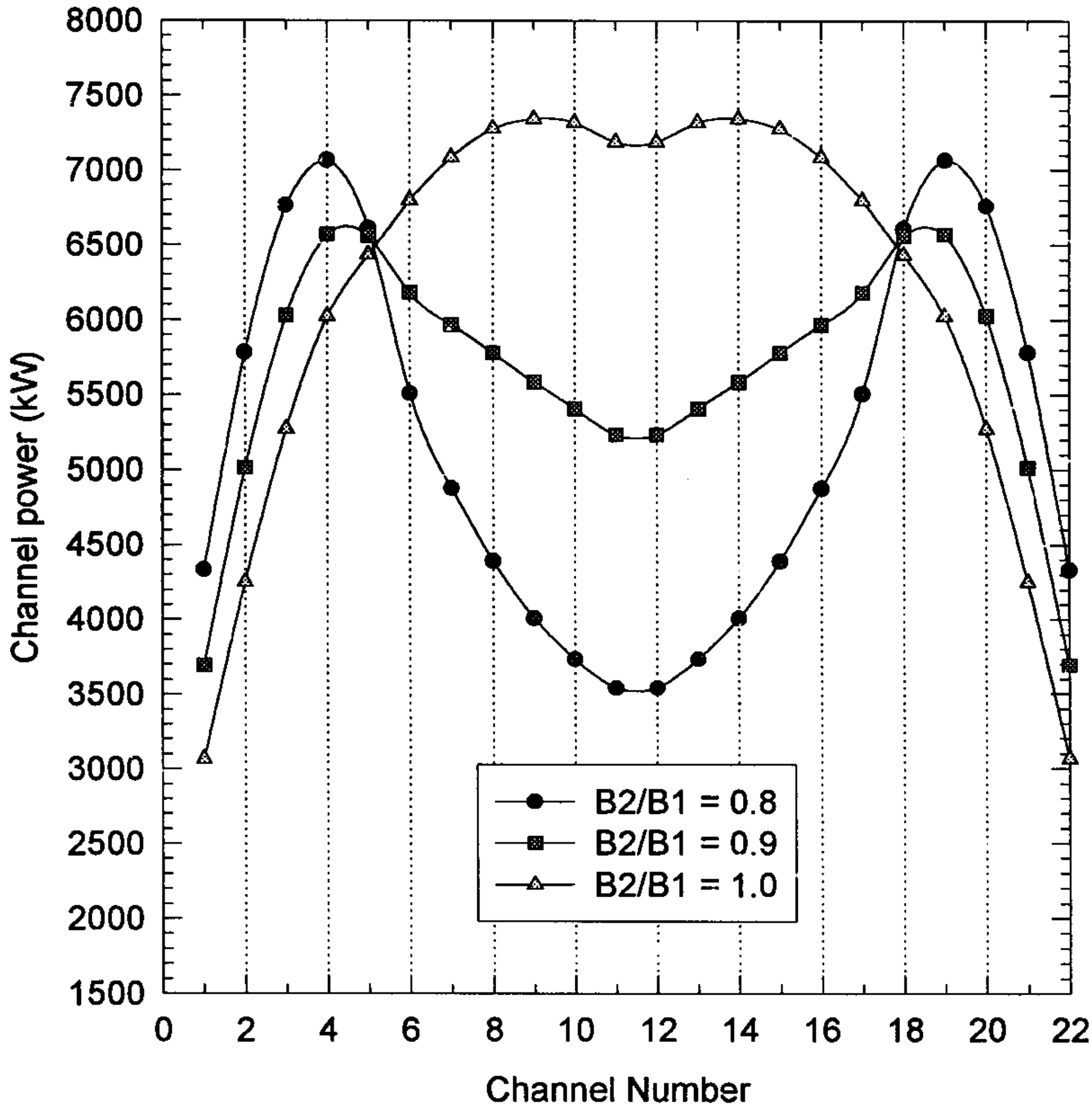


- reduction of adjuster worth: radial flattenning, more absorption in fuel
- evaluation of CPPF: age model

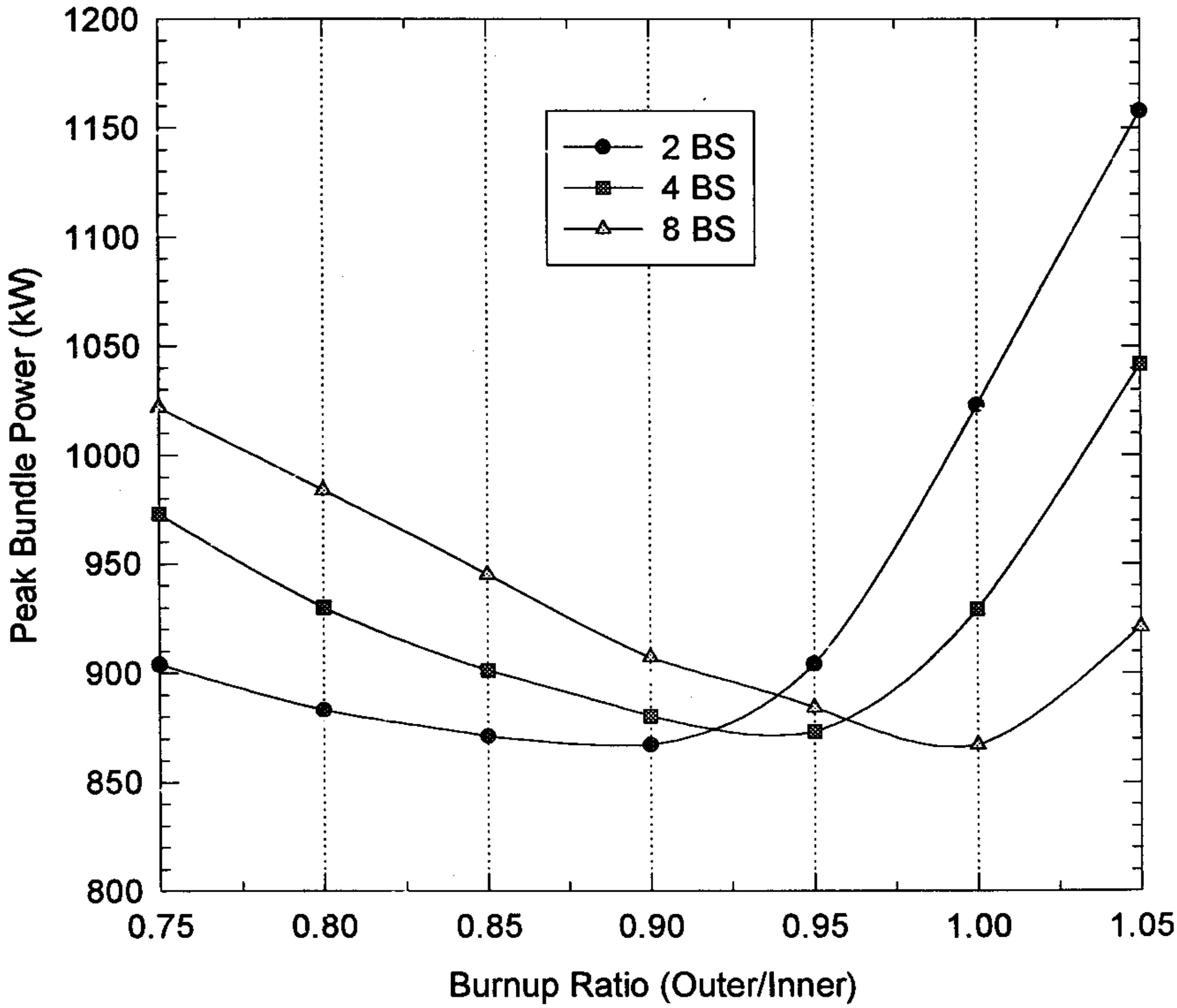
D. Rozon
December 1997



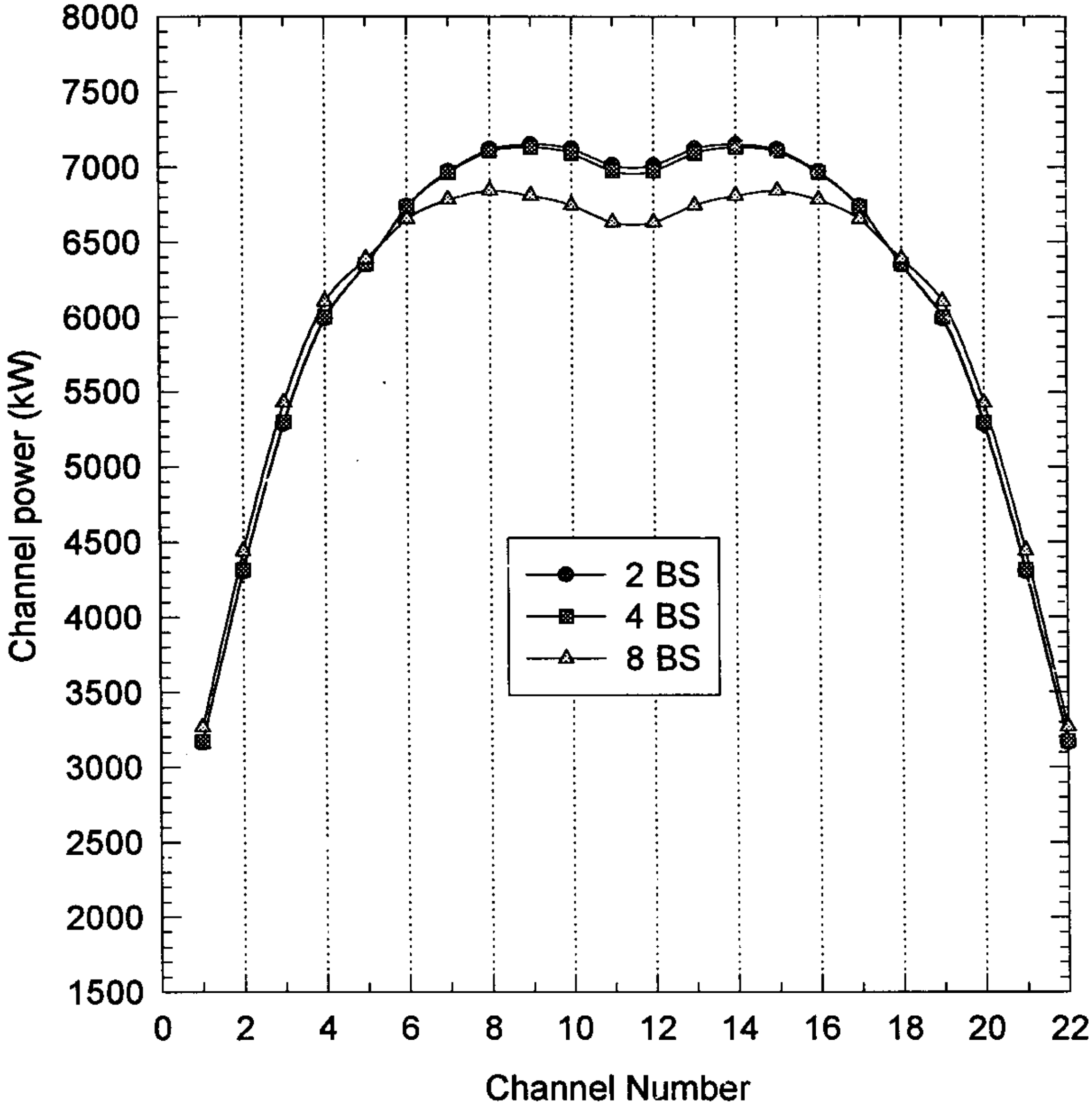
Radial Channel Power Profiles
(NAT. U, 8 BS, Row L)



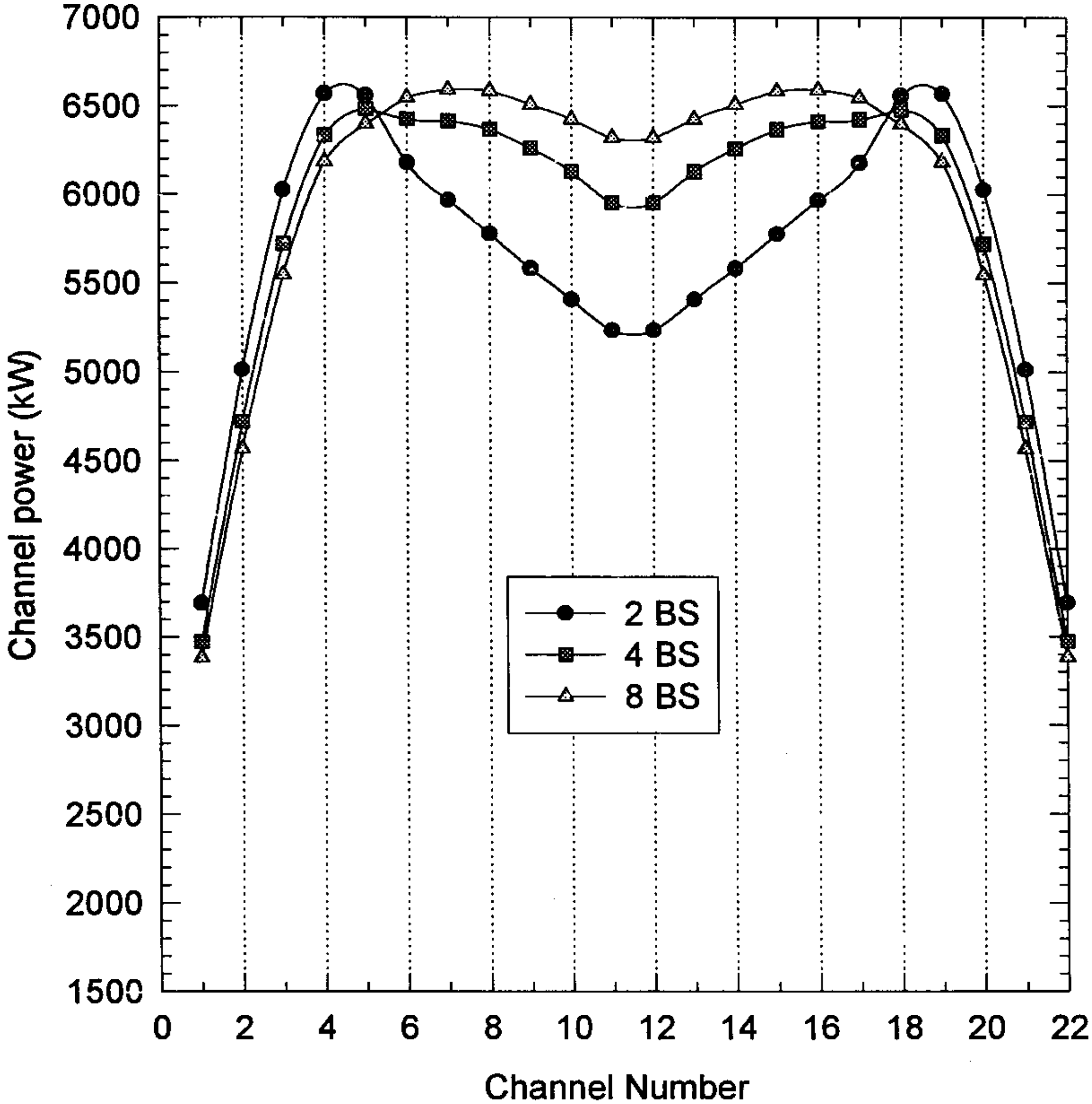
Radial Channel Power Profiles
(SEU, 1.2 w/o, 2 BS, Row L)



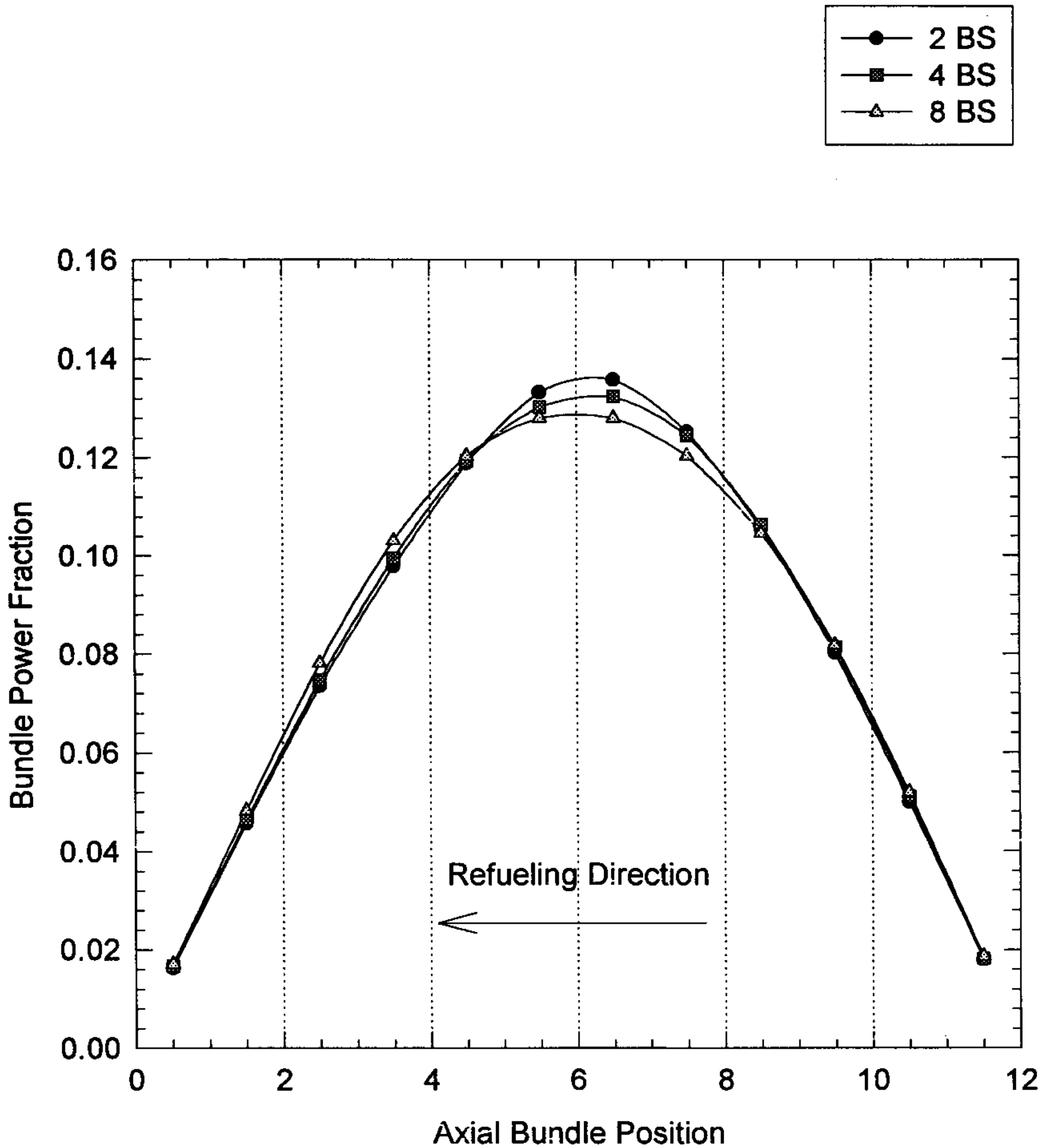
Peak Bundle Power for Each Refueling Scheme
(SEU, 1.2 w/o)



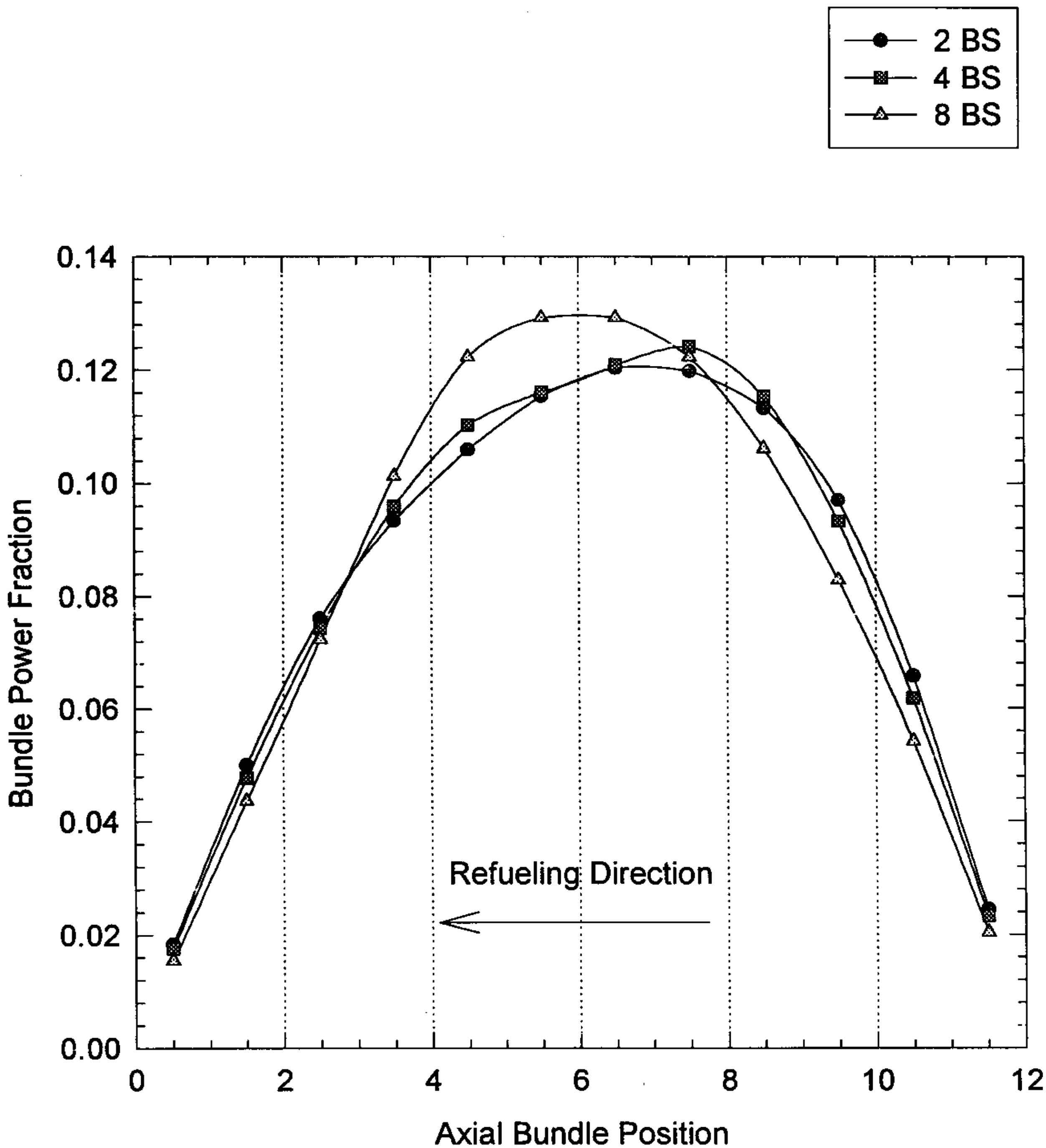
Radial Channel Power Profiles
(NAT. U, Row L)



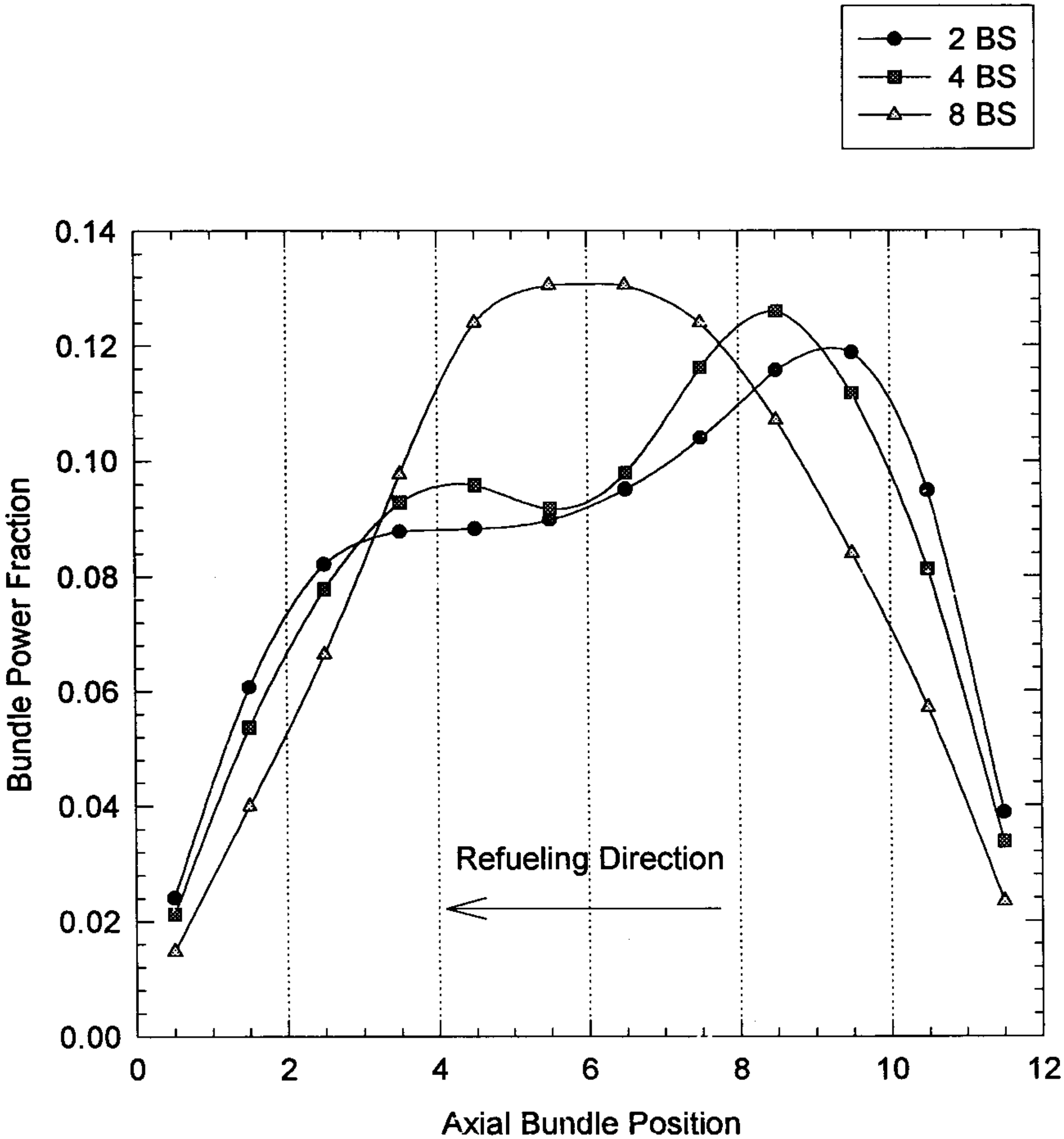
Radial Channel Power Profiles
(SEU, 1.2 w/o, Row L)



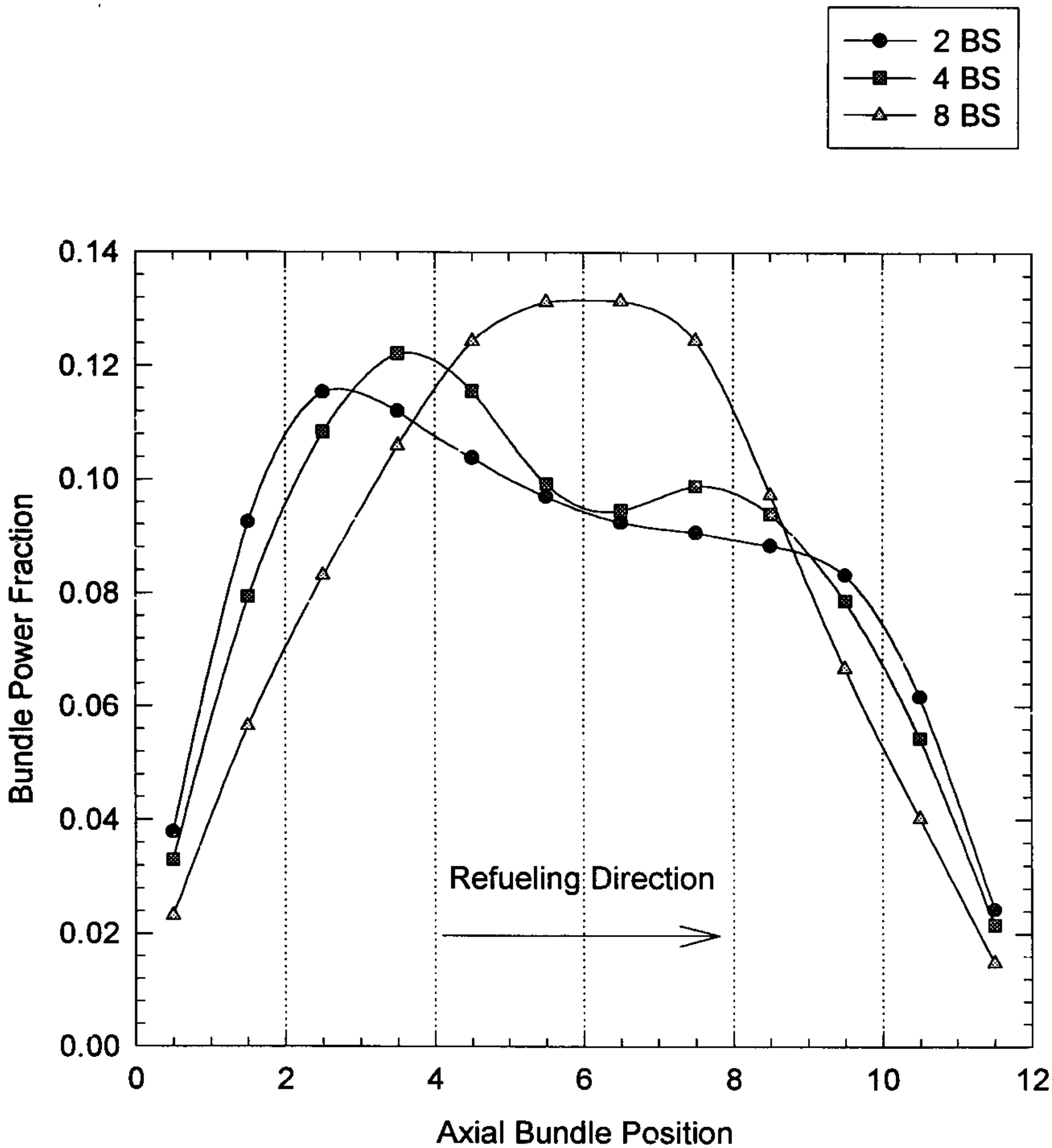
Typical Time-Average Axial Power Profiles
(NAT. U, Channel L3)



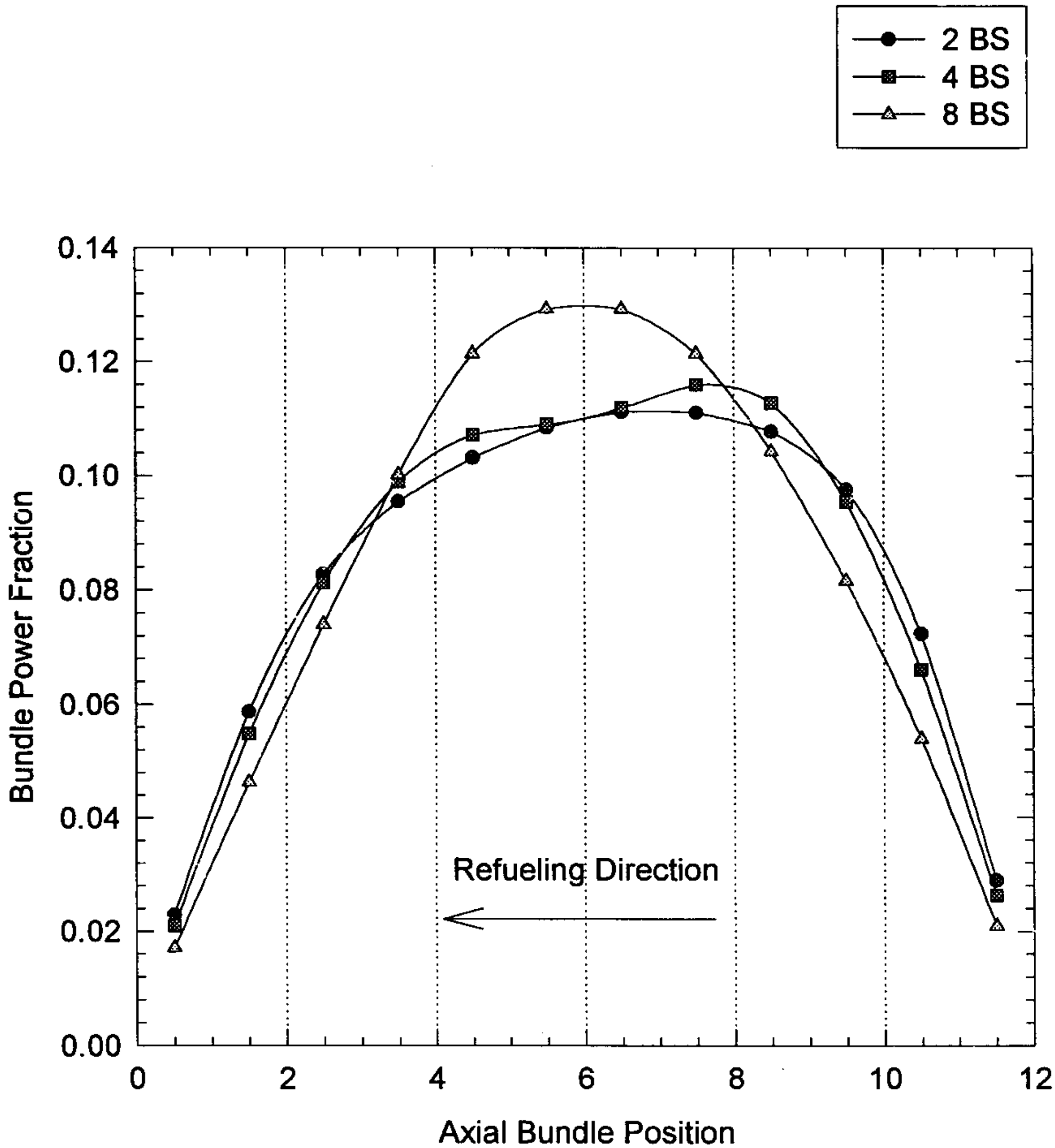
Typical Time-Average Axial Power Profiles
(SEU, 0.9 w/o, Channel L3)



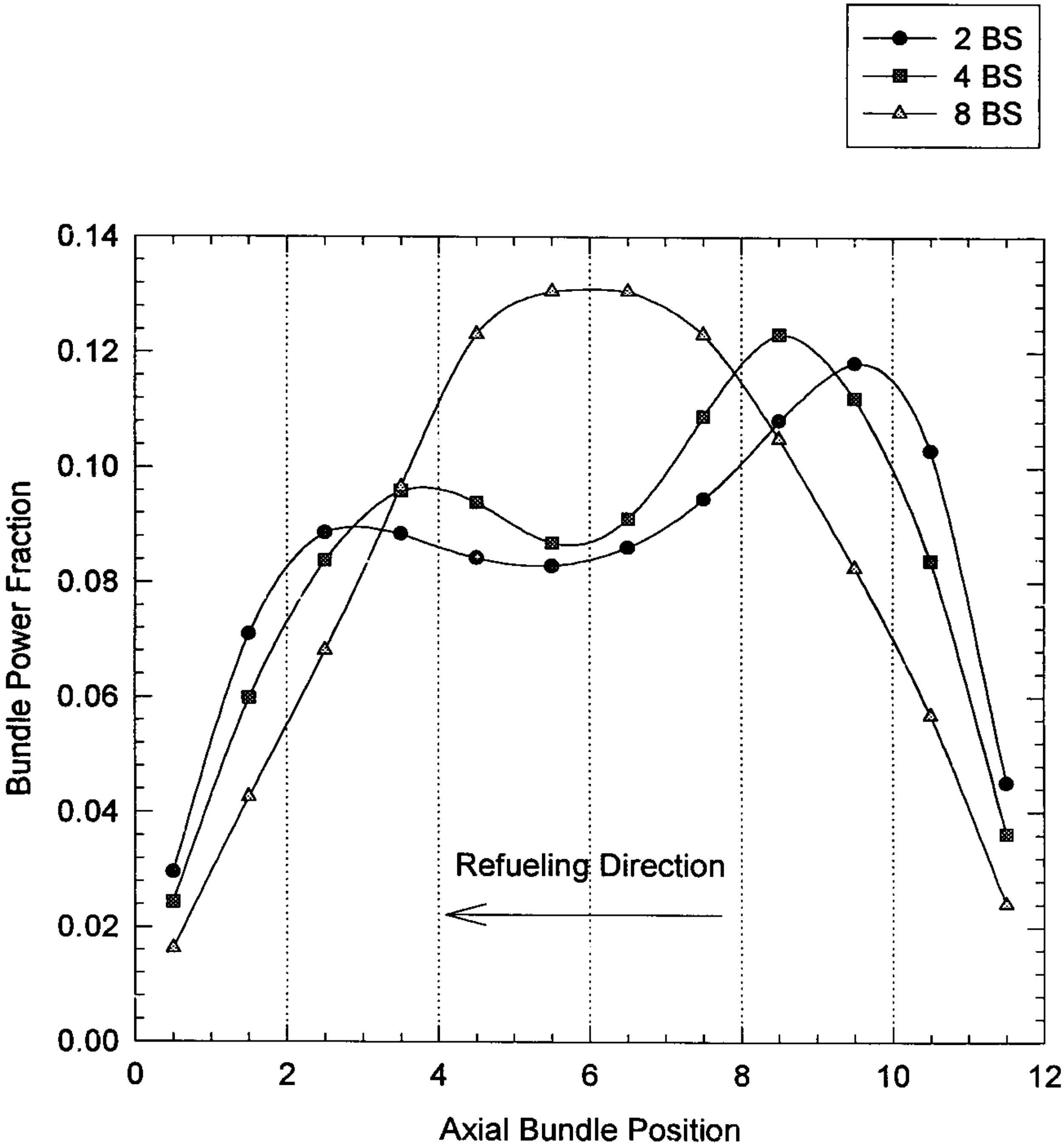
Typical Time-Average Axial Power Profiles
(SEU, 1.2 w/o, Channel L3)



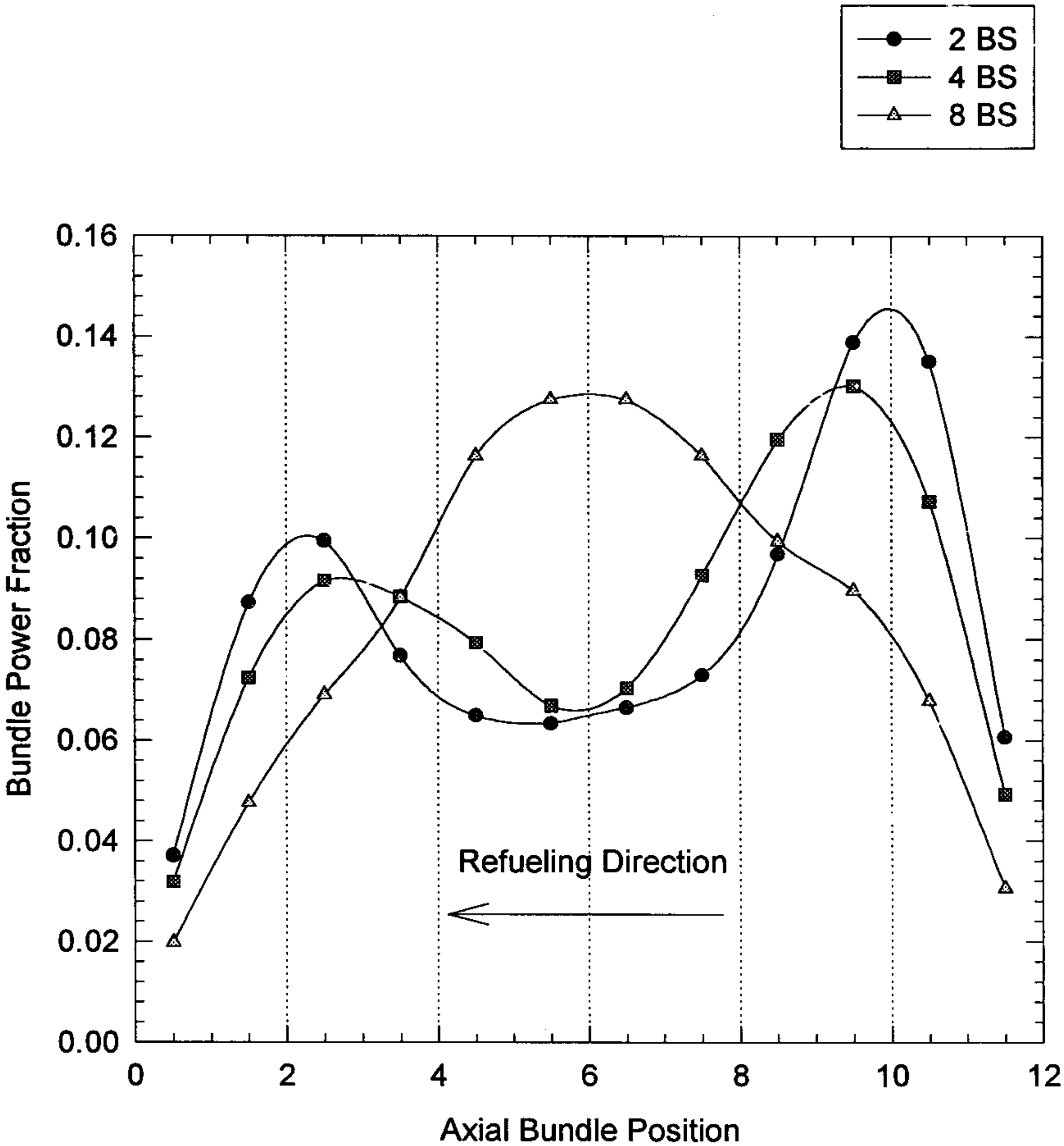
Typical Time-Average Axial Power Profiles
(SEU, 1.2 w/o, Channel T16)



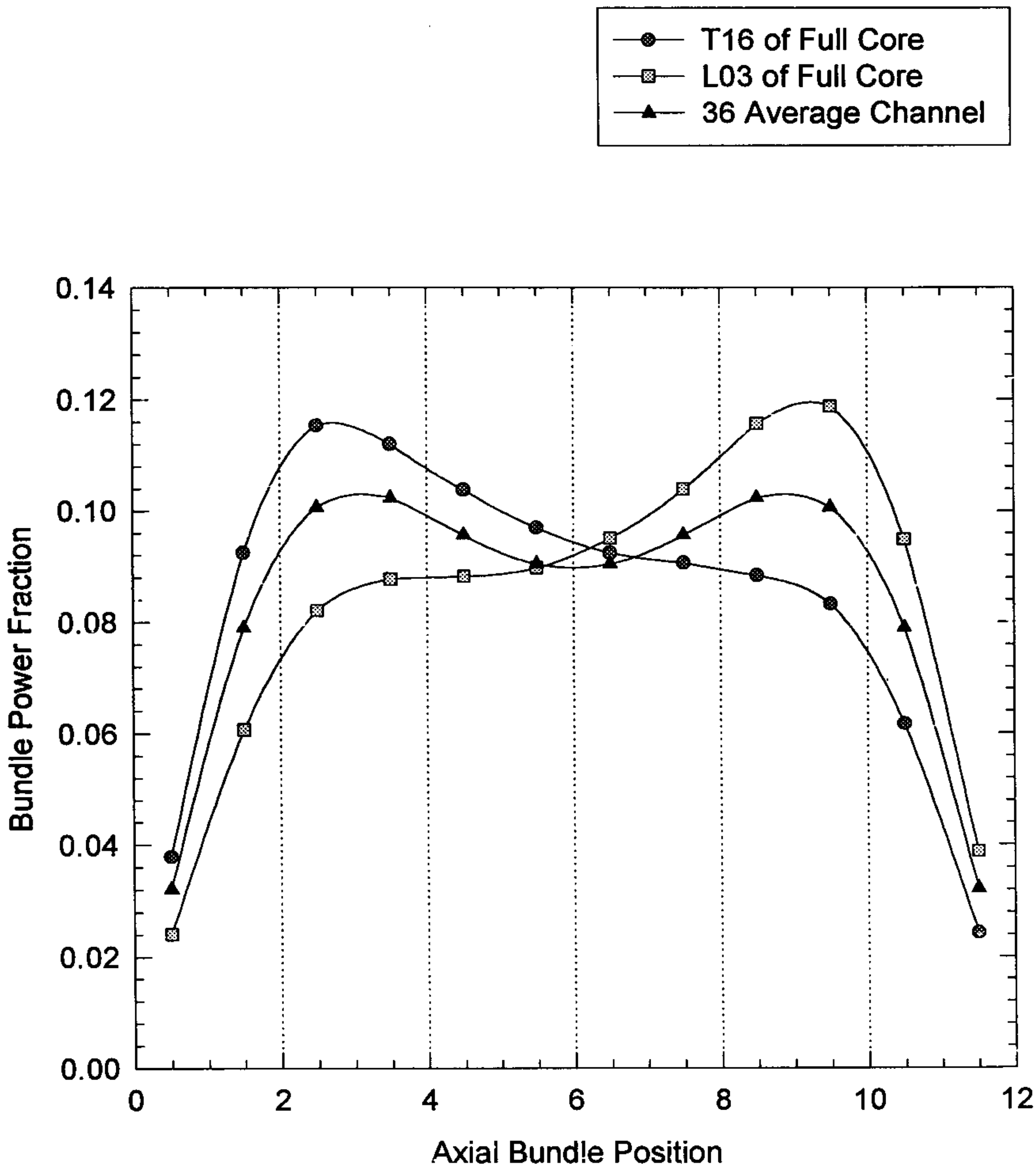
Typical Time-Average Axial Power Profiles
(MOX, 0.9 w/o, Channel L3)



Typical Time-Average Axial Power Profiles
(MOX, 1.2 w/o, Channel L3)



Typical Time-Average Axial Power Profiles (SEU, 1.2 w/o, Channel L11)



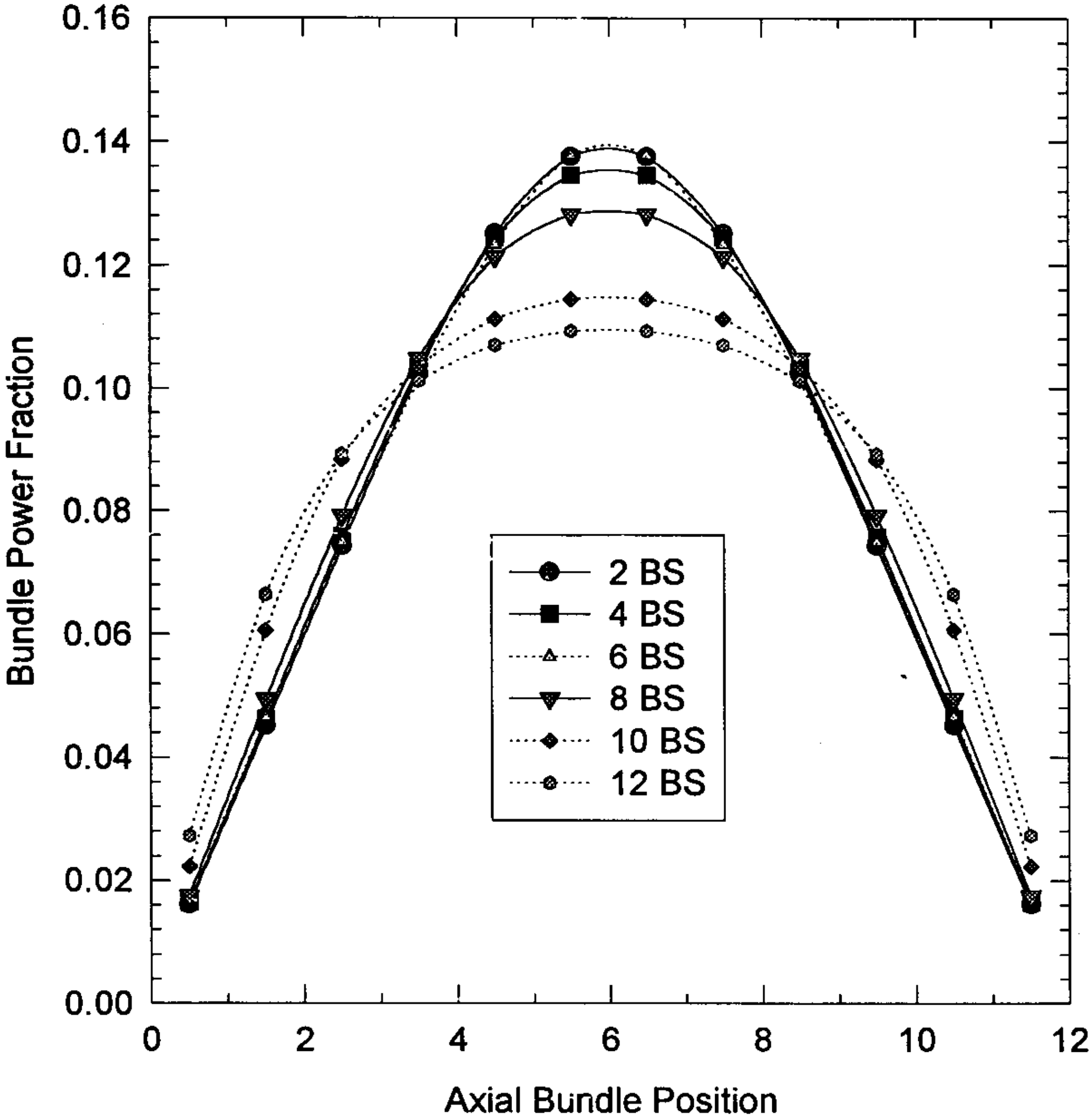
Typical Time-Average Axial Power Profiles
(SEU, 1.2 w/o, 2 BS)

36-Channel Supercell Model

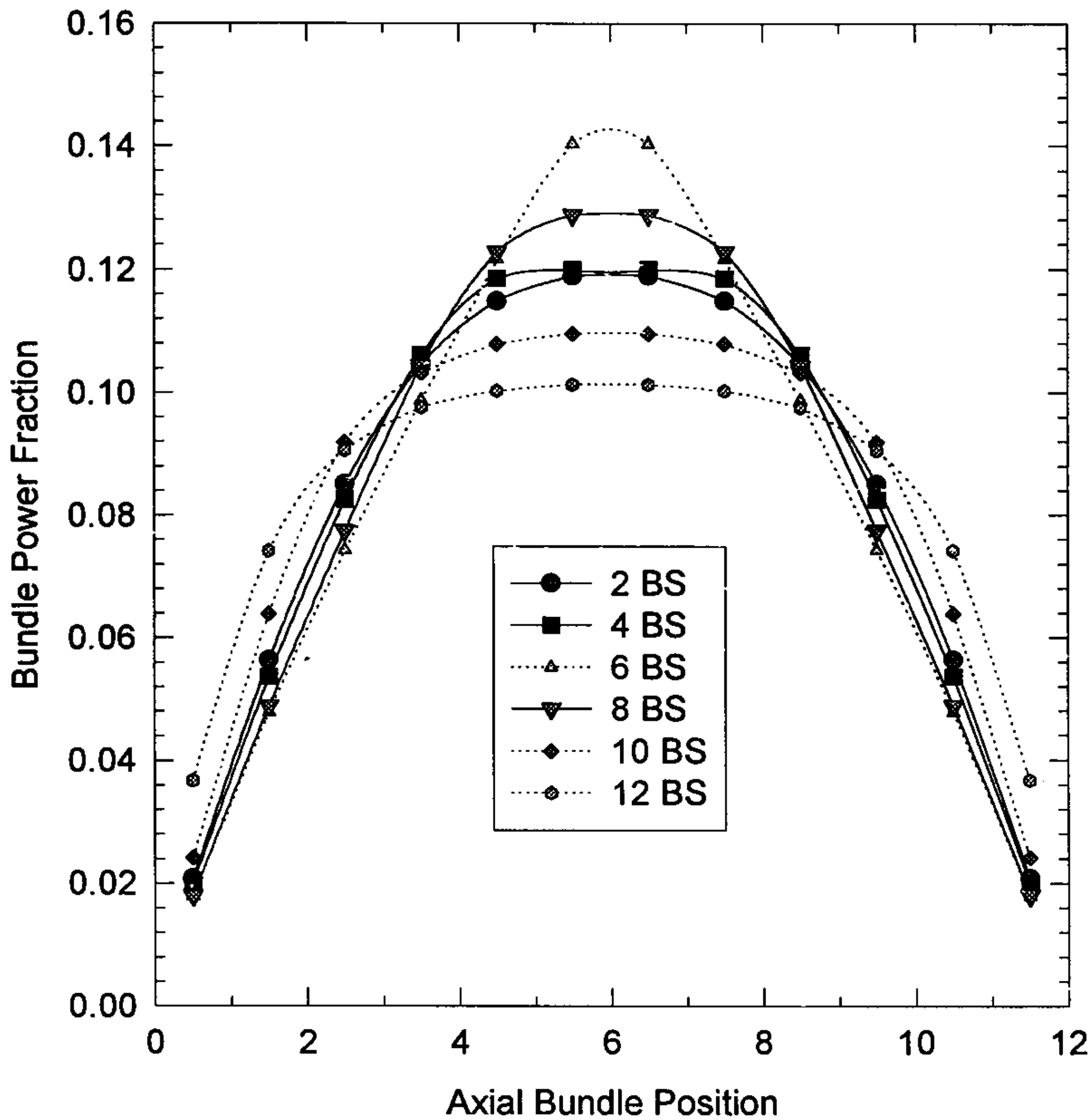
- symmetry
  - o infinite lattice: periodic boundary conditions in X-Y plane
  - o no symmetry in Z-direction (12 bundles in channels)
- bi-directional fueling schemes
  - o explicit representation of axial fuel movements in each channel
  - o alternating fueling direction in neighboring channels
  - o T/A and instantaneous calculations
- refueling sequence used to fix channel «age»: $f_j = n_j / 36$

- 22	+ 29	- 8	+ 17	- 14	+ 27
+ 7	- 16	+ 23	- 34	+ 9	- 4
- 30	+ 11	- 28	+ 19	- 24	+ 35
+ 3	- 20	+ 33	- 2	+ 21	- 6
- 12	+ 25	- 10	+ 15	- 26	+ 31
+ 5	- 36	+ 13	- 32	+ 1	- 18

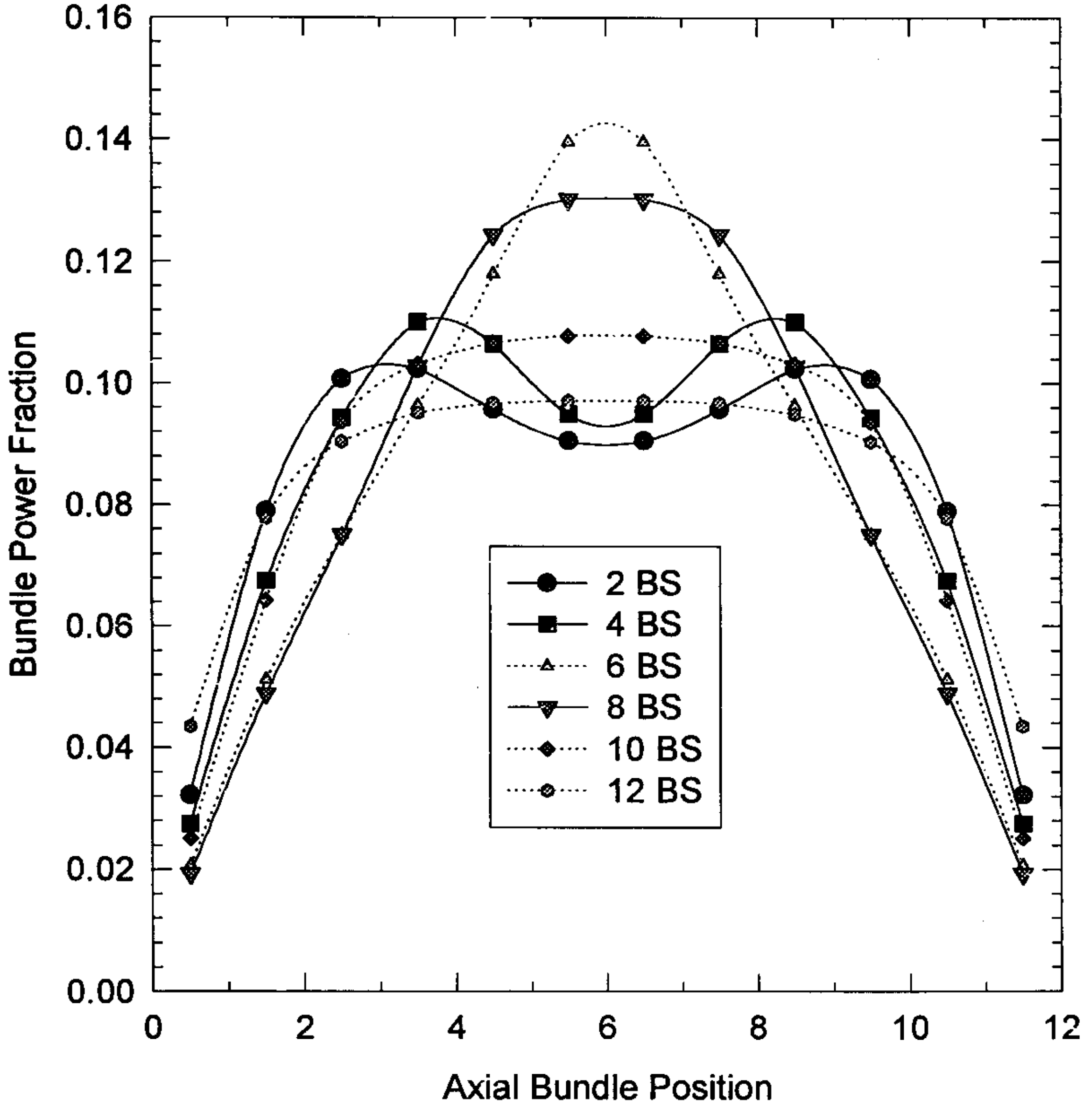
periodic boundary conditions
in X-Y plane



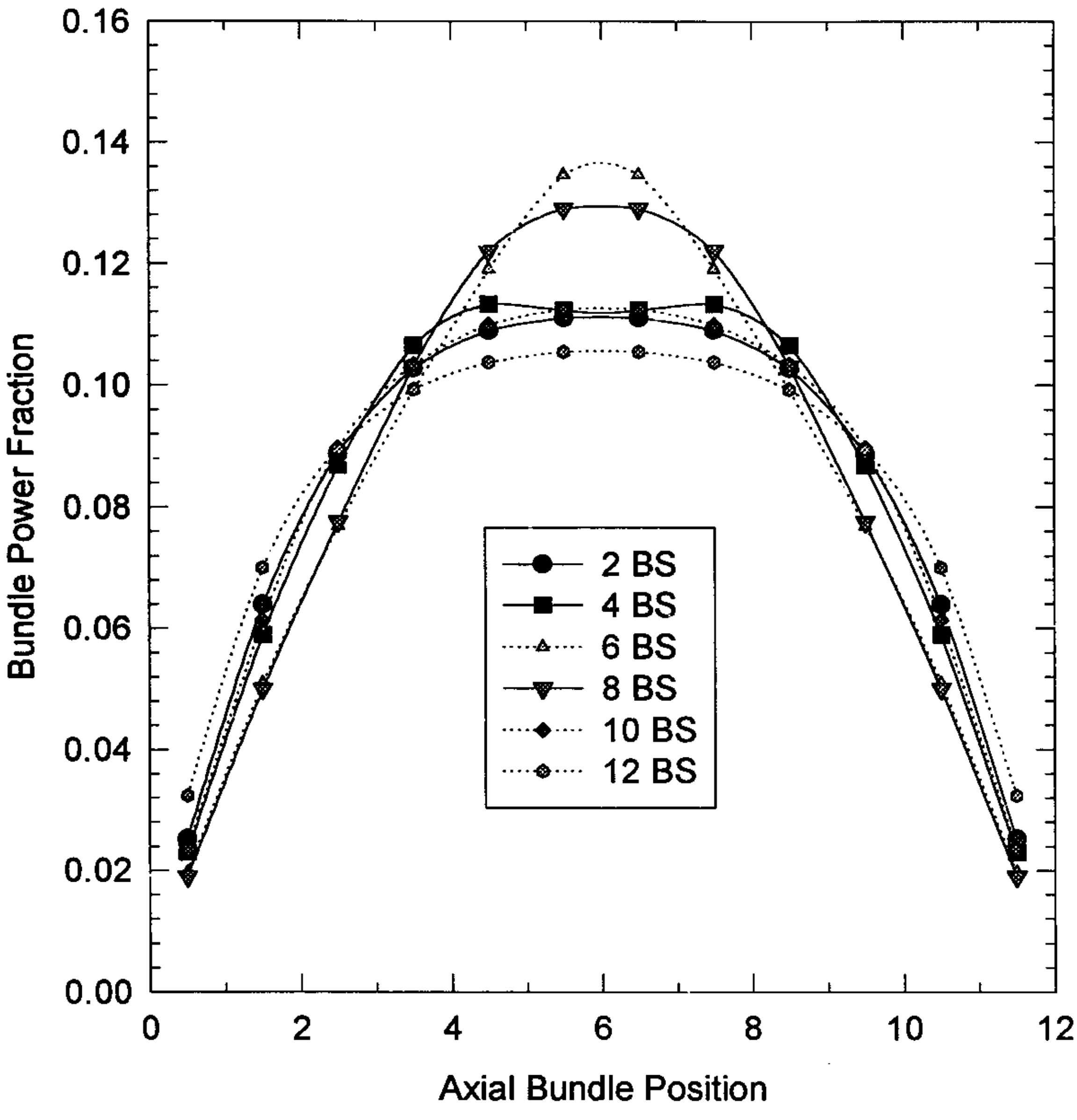
Bundle Powers in an Average Channel with Various Bundle Shift Strategies (NAT. U)



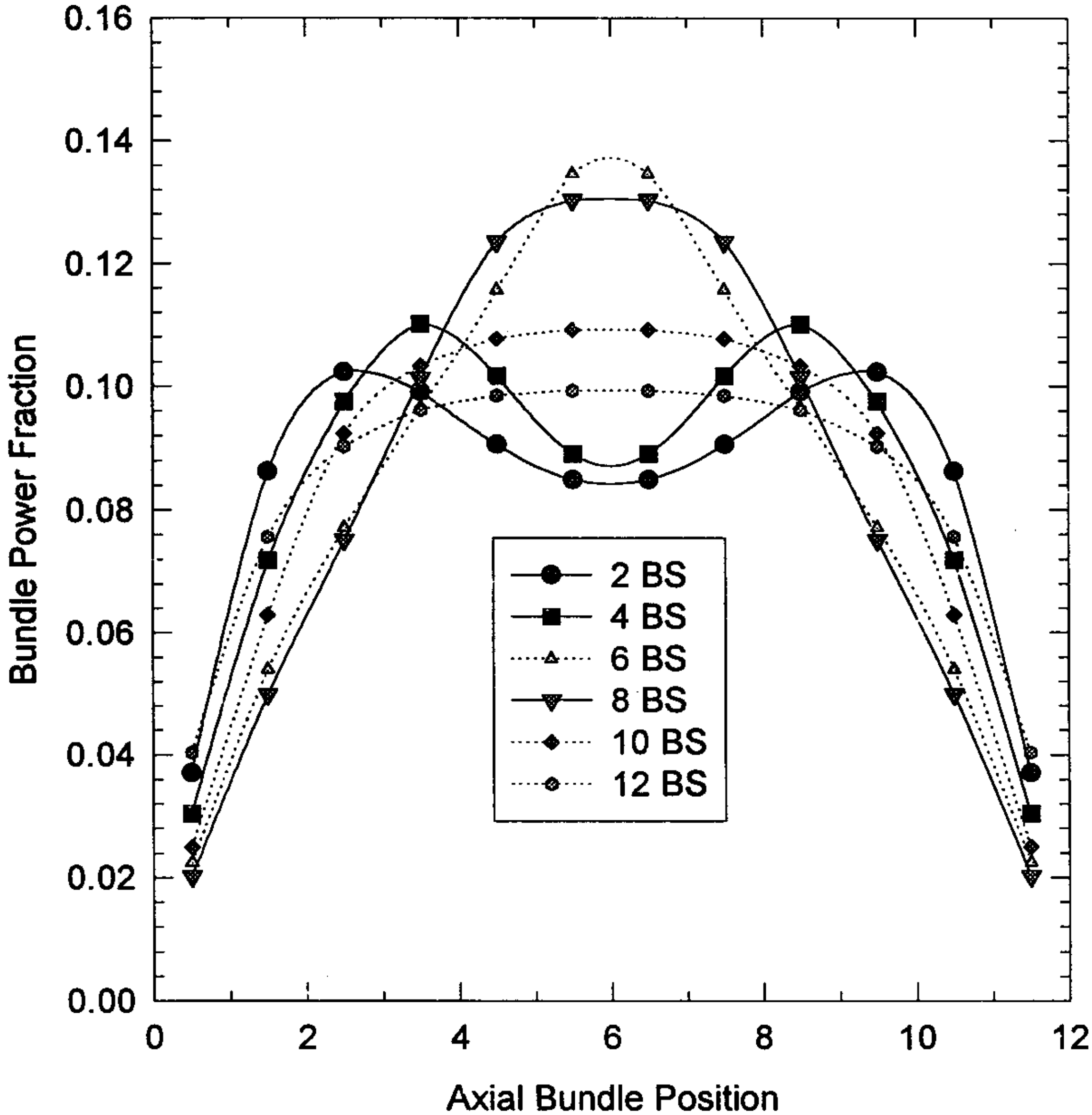
Bundle Powers in an Average Channel with Various Bundle Shift Strategies (SEU, 0.9 w/o)



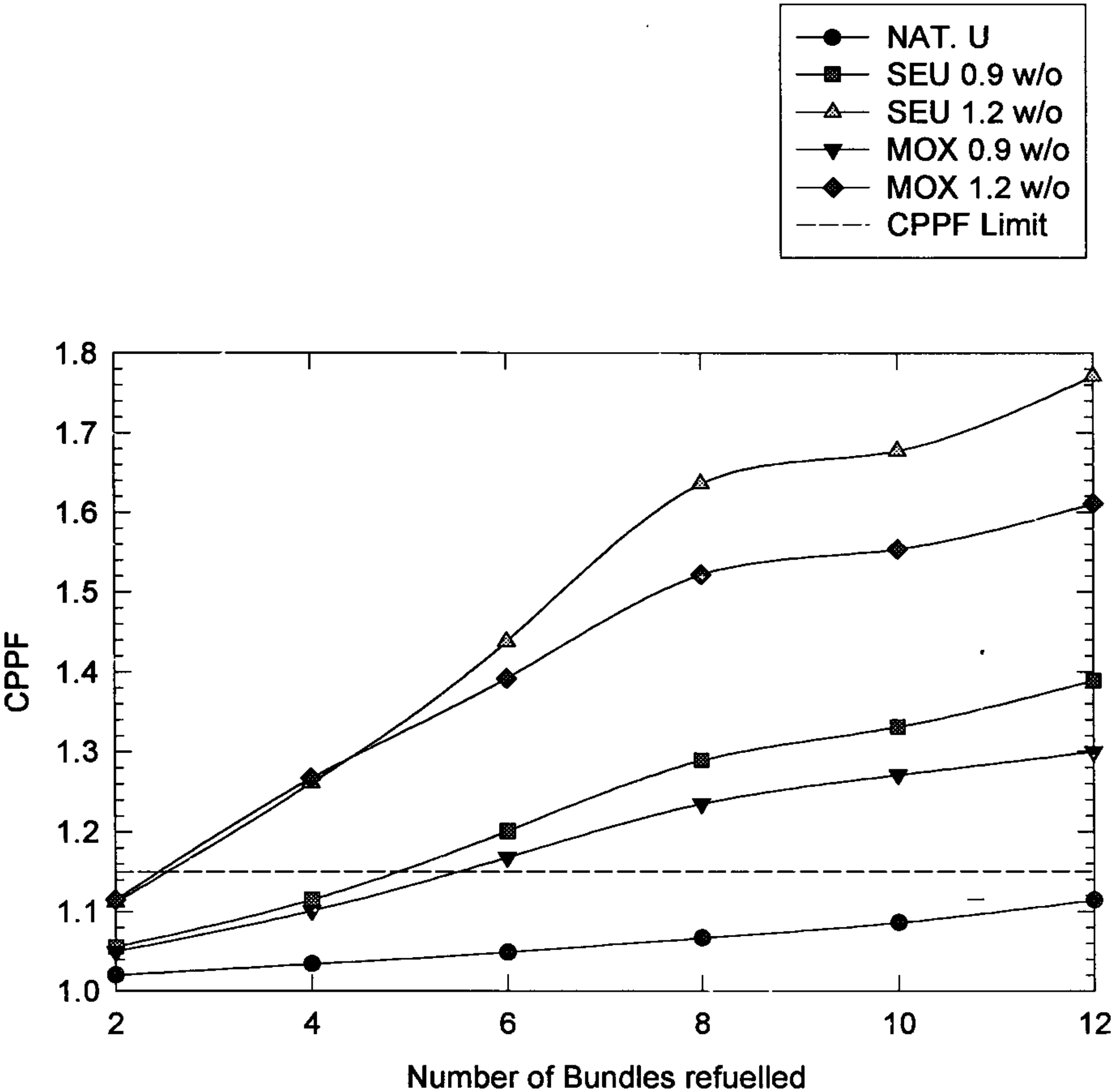
Bundle Powers in an Average Channel with Various Bundle Shift Strategies (SEU, 1.2 w/o)



Bundle Powers in an Average Channel with Various Bundle Shift Strategies (MOX, 0.9 w/o)



Bundle Powers in an Average Channel with Various Bundle Shift Strategies (MOX, 1.2 w/o)



Channel Power Peaking Factors(CPPF)
vs. Bundle Shift Strategy
(36 Full Channels Caculation)

Refueling Sequence for Age Map in DONJON

Block Sequence

```
 7 11  1  9
13  3 15  5
 6 16  4 14
10  2 12  8
```

Refueling Sequence in odd blocks

```
22 29  8 17 14 27
 7 16 23 34  9  4
30 11 28 19 24 35
 3 20 33  2 21  6
12 25 10 15 26 31
 5 36 13 32  1 18
```

Refueling Sequence in even blocks

```
21 28  7 16 13 26
 6 15 22 33  8  3
29 10 27 18 23 34
 2 19 32  1 20  5
11 24  9 14 25 30
 4 35 12 31 36 17
```

Refueling Sequence in the Whole Core

```
                                 358  91  39  67 162 234
                     315 113 294 196 248 366 309 109 287 191 245 363
                  59  33 208 346  19 219  61  25 202 340  14 213  55  31
             271 322 123 261 103 156 273 323 118 254  97 149 265 318 122 260
         333   7 184  51 377 135 334   8 185  44 371 127 327   1 179  50 376 133

         176 145 283 226 301  79 173 142 279 232 307  86 177 147 285 228 303  81
     241 360  93  40  69 164 236 353  88  37  76 171 243 361  95  42  71 166 238 355
     296 198 250 367 311 110 289 193 246 364 316 116 298 200 252 369 313 112 291 195
 210 348  21 221  63  27 204 342  16 215  57  35 211 350  23 223  65  29 206 344  18 217
 262 105 158 275 324 119 256  99 151 267 319 125 263 107 160 276 325 121 258 101 153 269
 378 137 336  10 187  46 372 129 329   3 181  53 379 139 338  12 189  48 374 131 331   5

 292  72 167 132 270 224 299  77 172 140 277 216 290  70 165 130 268 222 297  75 170 138
 154 229 345  82  30  66 161 233 351  87  36  58 152 227 343  80  28  64 159 231 349  85
 102 281 183 239 356 308 108 286 190 244 362 302 100 280 182 237 354 306 106 284 188 242
     332   6 207  49  24 201 339  13 212  54  17 194 330   4 205  47  22 199 337  11
      90 144 259 314 117 253  96 148 264 317 111 247  89 143 257 312 115 251  94 146
         321 375 175  43 370 126 326 380 178  38 365 120 320 373 174  41 368 124

         168 134 272 214 288  68 163 128 266 220 295  74 169 136 274 218 293  73
              83  32  56 150 225 341  78  26  62 157 230 347  84  34  60 155
                 357 300  98 278 180 235 352 305 104 282 186 240 359 304
                      15 192 328   2 203  45  20 197 335   9 209  52
                                 141 255 310 114 249  92
```

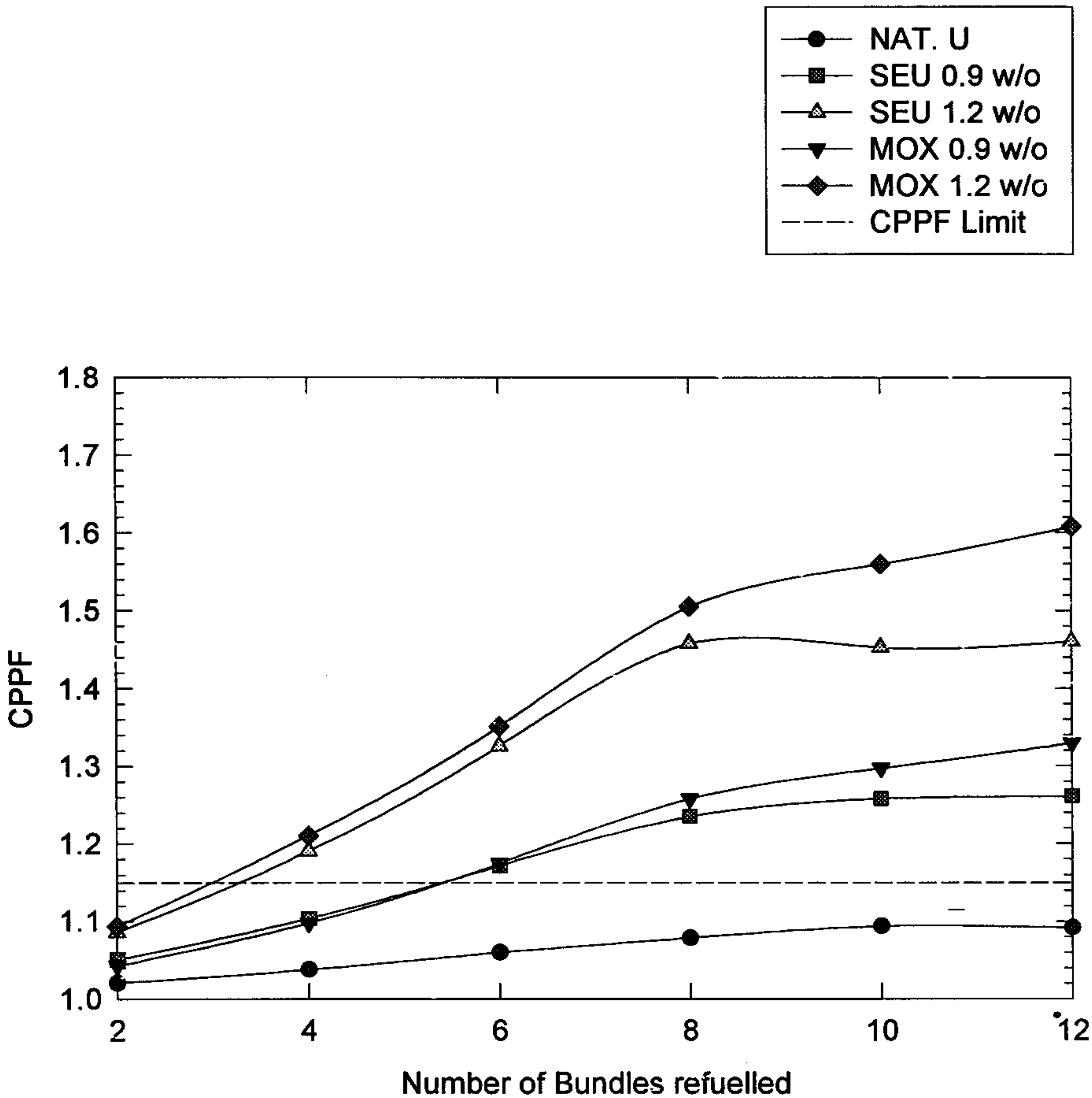
CPPF

FUEL TYPE	*BS*	*AVERAGE EXIT BURNUP MWd/T(U)*	***CPPF***
Nat. U	2	7710	1.021 (P18)
	4	7668	1.038(V12)
	8	7613	1.079 (E16)
SEU 0.9 w/o	2	14226	1.051 (P18)
	4	14292	1.104 (E16)
	8	14768	1.235 (E16)
SEU 1.2 w/o	2	21715	1.086 (L10)
	4	22169	1.191 (E16)
	8	23799	1.458 (P18)
MOX 0.9 w/o	2	5931	1.043(N17)
	4	6051	1.098 (E16)
	8	6327	1.258 (E16)
MOX 1.2 w/o	2	12454	1.094 (L10)
	4	12796	1.211 (E16)
	8	13907	1.505 (E16)

- *CPPF* increases significantly with enrichment (ε) and decreases when bundle shift (n) is reduced:

  $\Rightarrow$ 2 BS is required for enrichement above 1%

- *axial flux shape* is significantly affected by combined effect of *bi-directional* bundle scheme and enrichment:
  - o 36 channel supercell can be used to find correlation between CPPF and (ε, n, B) $\Rightarrow$ optimization with T/A model alone (ex. OPTEX-4)
  - o optimization of refueling sequence within blocks for age pattern in full core calculations



Channel Power Peaking Factors(CPPF)
vs. Bundle Shift Strategy
(Full Core Caculation)

6. DUPIC

- Direct use of PWR fuel in CANDU:
  - o spent fuel in PWR contains ≈ 1.5 g/kg of fissile material (U+Pu):
    - ⇒ chemical reprocessing for recycle of U and Pu in LWR or FBR (Europe, Japan, ...)
    - ⇒ remove sheath and create new powder for resintering and remote fabrication of CANDU fuel bundle

- Questions:
  1. What burnup can be acheived in CANDU using PWR fuel after a 10 y cooldown period?
  2. Is performance of DUPIC fuel in CANDU only a function of the burnup of the spent PWR assemblies?
  3. Sensitivity of DUPIC performance characteristics to the uncertainty in initial nuclide field.

- DRAGON/DONJON analysis of DUPIC:[1]

  PWR 900 (Daya Bay) ⇒ CANDU 6 (CANFLEX fuel)

  - o same ENDF-B5 library used for both lattices (89 groups)
  - o reference (nominal 3.2% enriched replacement fuel)
    - ⇒ exit burnup: 30-35 GWd/TeU
  - o axial distribution of burnup in PWR assembly (local parameters)
  - o use of burnable poisons (PWR and CANDU)
  - o increase initial enrichment in PWR:
    3.2% (30 GWd/TeU) ⇒ 3.5% (35 GWd/TeU)

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[1] Calculations performed by Dr Shen Wei, post-doctoral fellow at IGN. Results are preliminary.

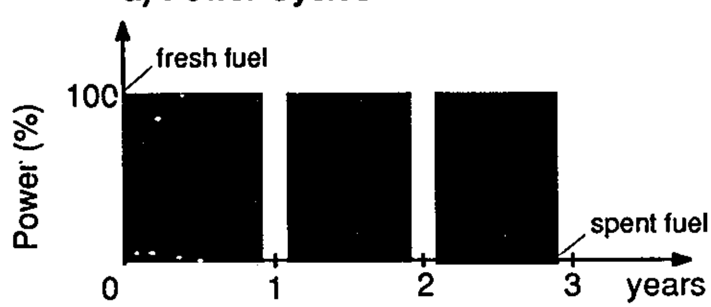
Generating Nuclide Fields of Spent PWR Fuel with DRAGON

- PWR cell/assembly
  - 17X17 assembly, water gap
  - 264 fuel pins + 24 guide tubes + 1 instrument thimble
  - nominal conditions:
    - o power density 39.95 kW/kg
    - o fuel density: 10.41 g/cc (460 kg/assembly)
    - o fuel temperature: 671.6 °C
    - o moderator temp.: 311.7 °C
  - Burnable Poison (BP) assembly:
    - o Boron-silicate glass, 16 or 12 BP rods

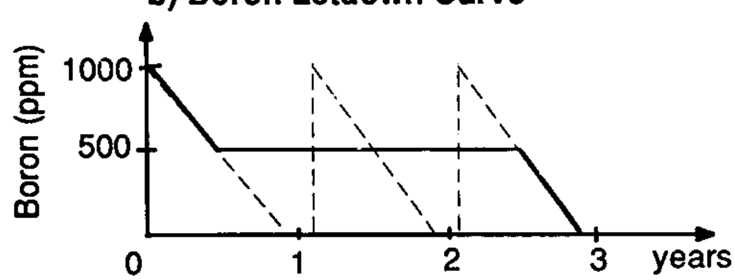
- Local Parameters

case	*power density*	*moderator temperature*	*fuel temperature*
top	80%	325 °C	615 °C
center	120%	312 °C	769 °C
bottom	80%	293 °C	583 °C
nominal	100%	312 °C	672 °C

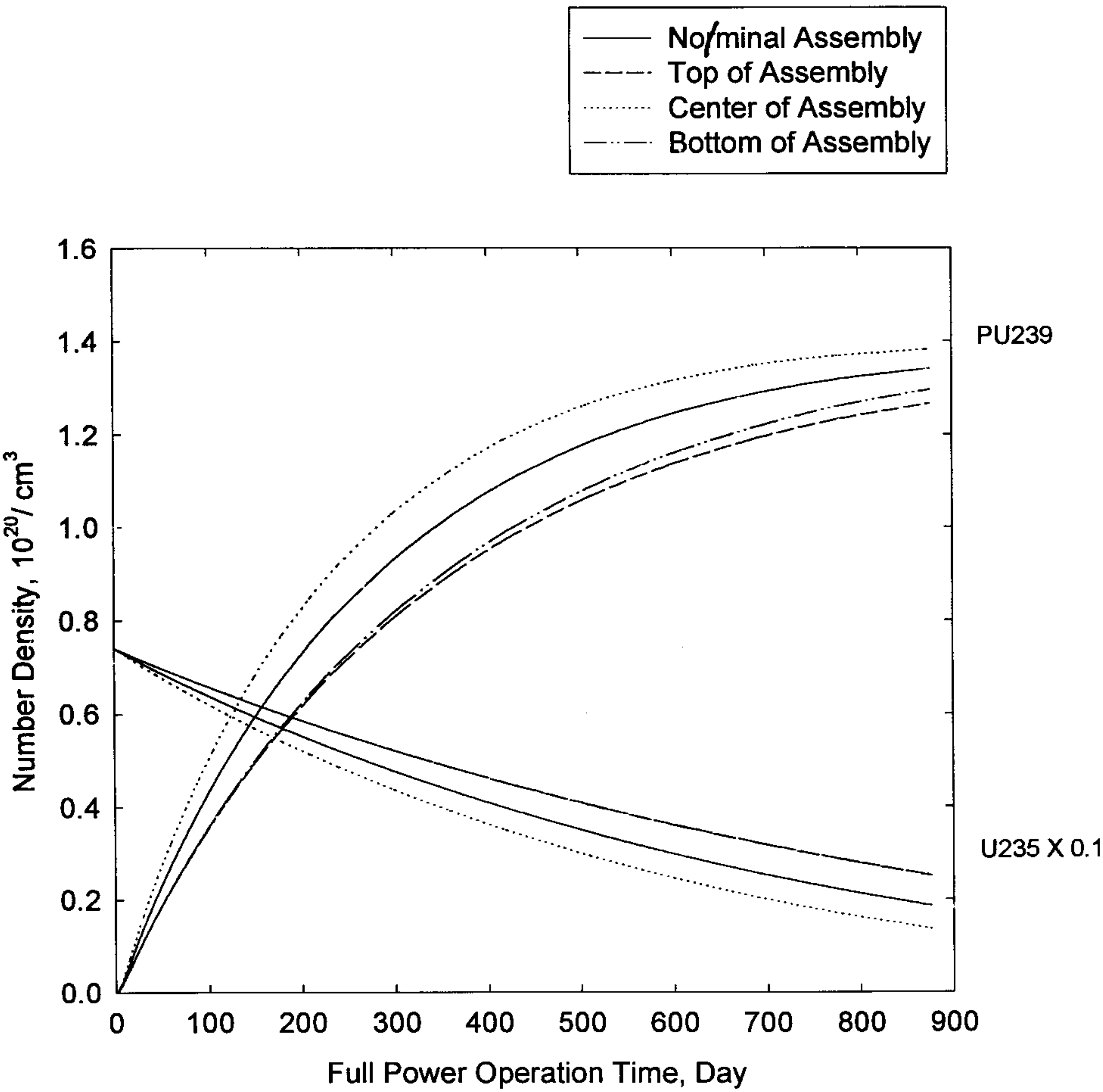
a) Power Cycles



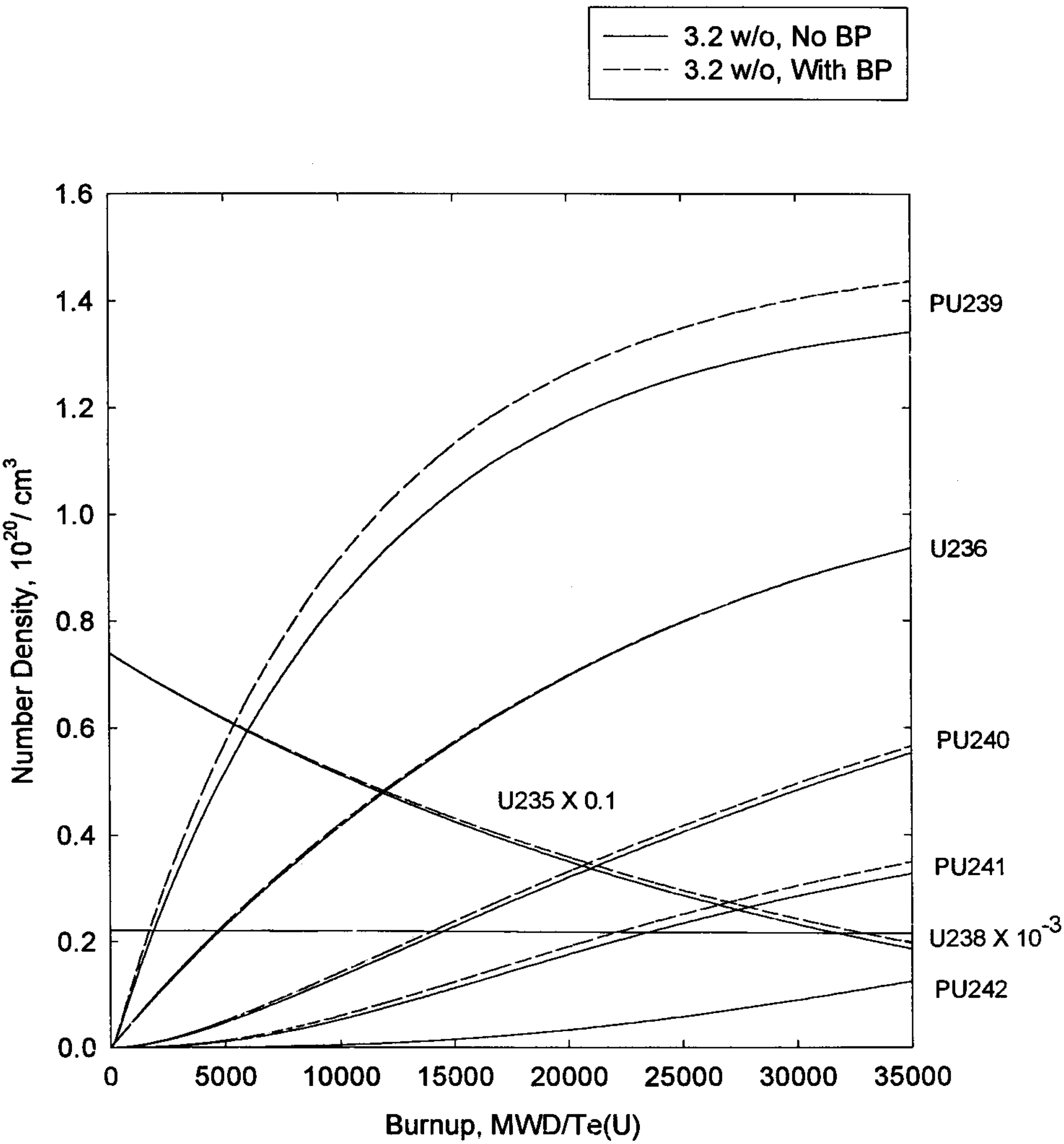
b) Boron Letdown Curve



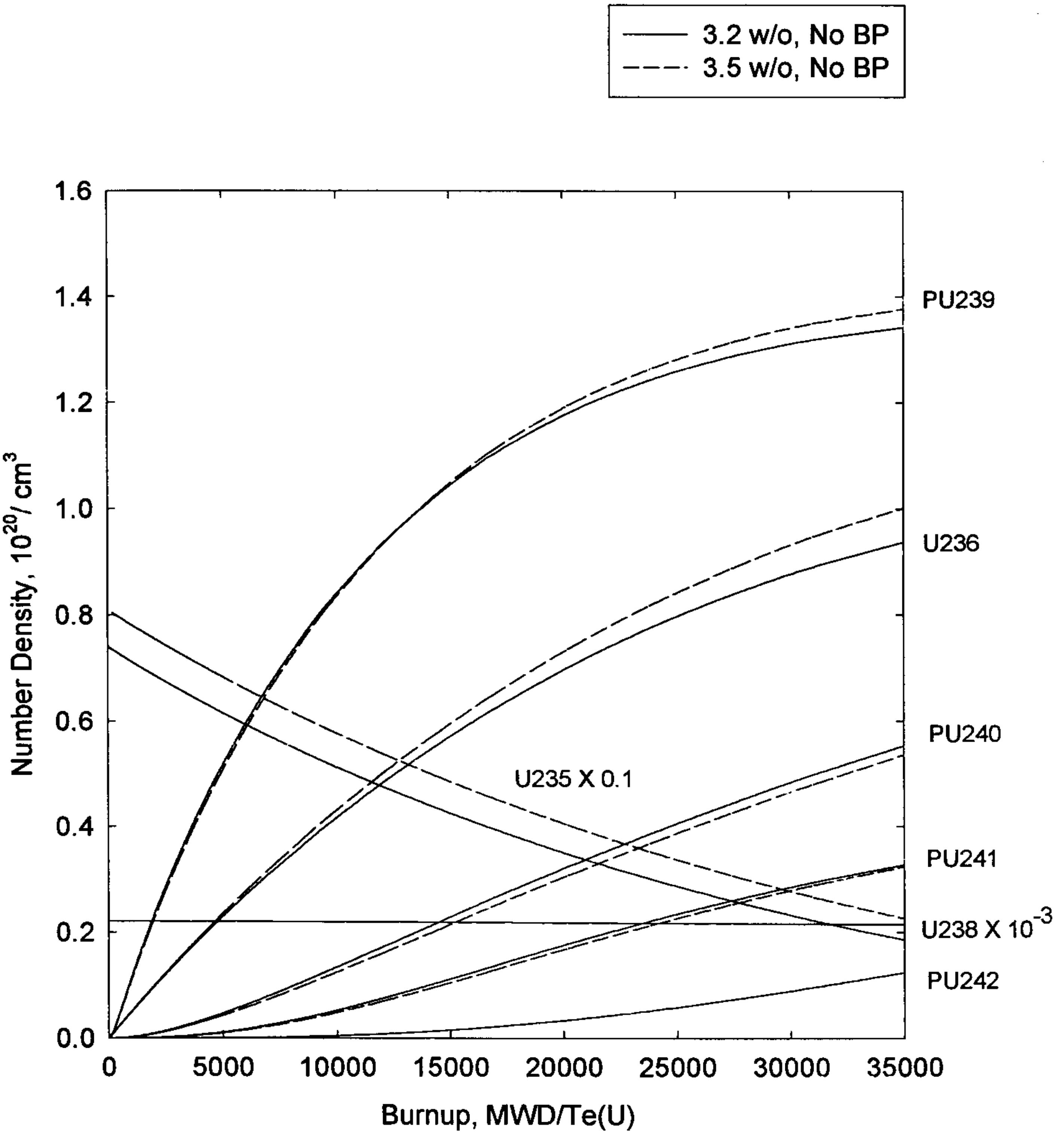
D. Rozon
December 1997



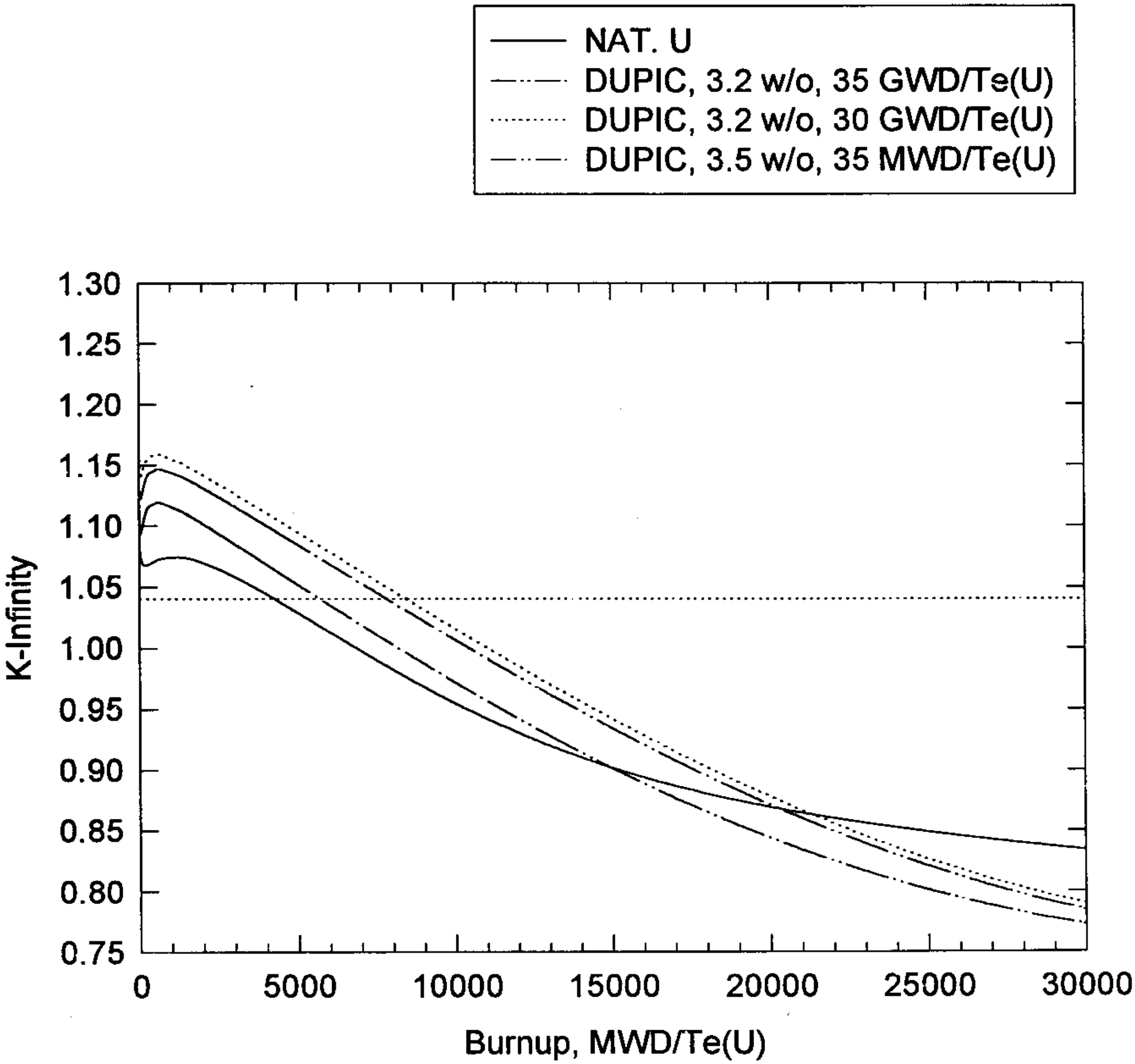
Variation of PWR Assembly Number Density versus Operation Time (3.2 w/o, No BP)



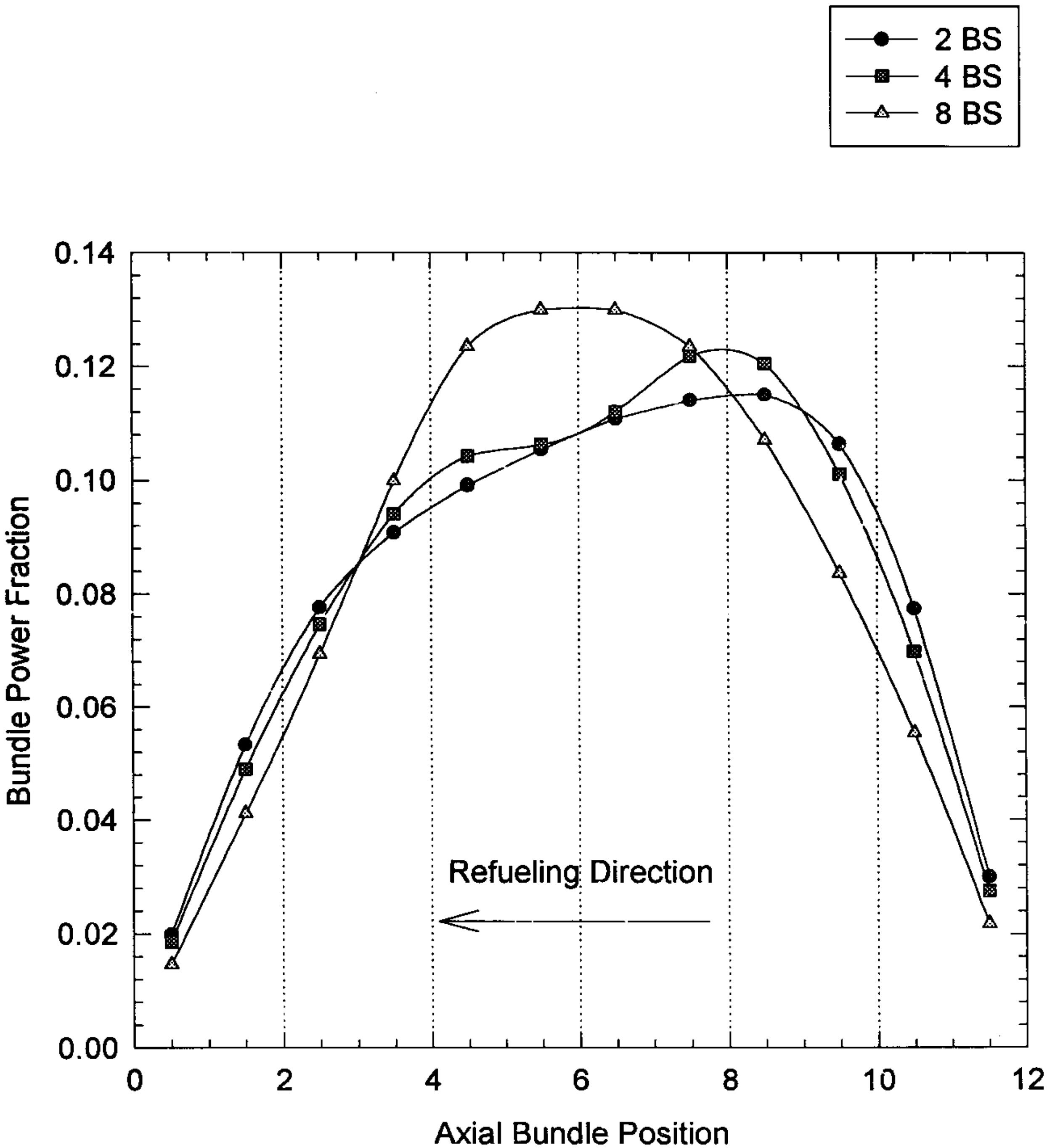
Variation of PWR Fuel Number Density versus Burnup



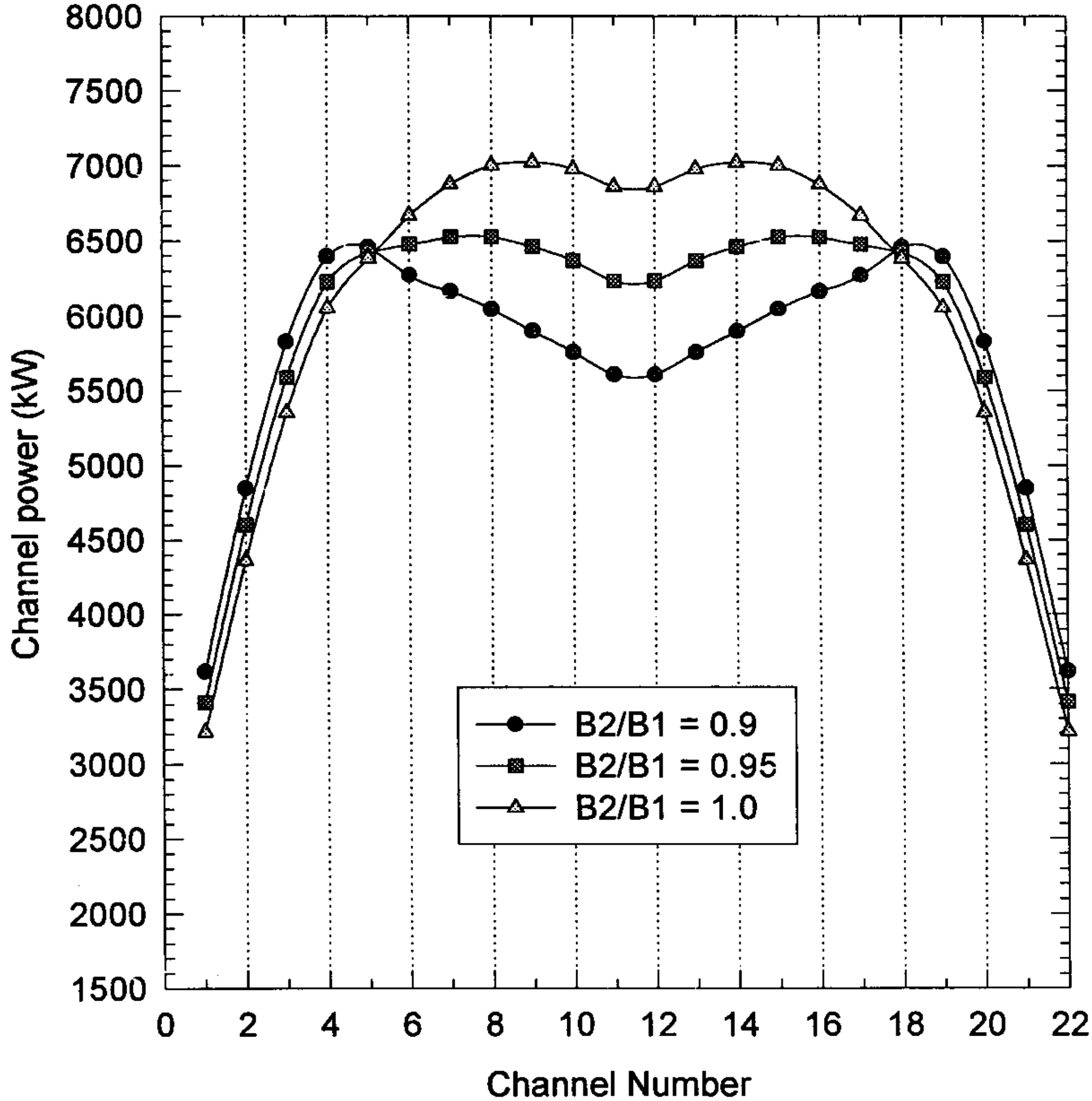
Variation of PWR Fuel Number Density versus Burnup



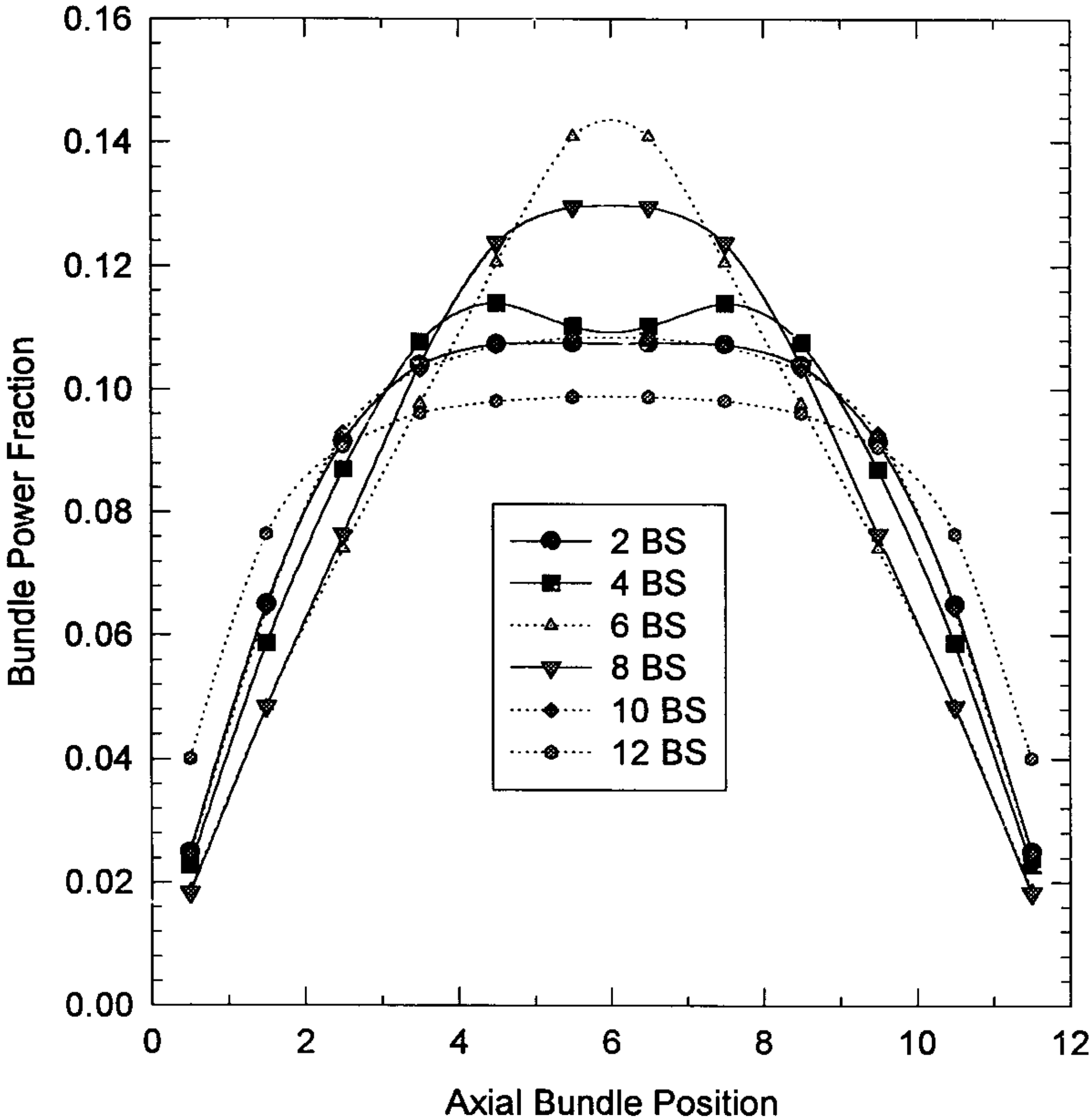
Lattice K-infinity vs. Burnup for Different Fuels



**Typical Time-Average Axial Power Profiles
(DUPIC, 3.2 w/o, 30 GWD/Te(U), Channel L3)**



Radial Channel Power Profiles
(DUPIC, 3.2 w/o, 35 GWD/Te(U), 2 BS, Row L)



Bundle Powers in an Average Channel with Various Bundle Shift Strategies (DUPIC, 3.2 w/o, 30 GWD/Te(U))

Numerical Results

Characteristics of DUPIC Core (CANFLEX, with BP)

SPENT PWR FUEL TYPE	BS	AVERAGE EXIT BURNUP MWd/T(U)	ADJUSTER WORTH (mk)
Nat. U	2	7710	17.49
	4	7668	17.62
	8	7613	16.64
3.2 w/o 35 GWD/Te(U)	2	11765	11.45
	4	11833	11.79
	8	12298	12.55
3.2 w/o 30 GWD/Te(U)	2	16702	9.72
	4	16970	11.40
	8	17934	12.80
3.5 w/o 35 GWD/Te(U)	2	15878	9.63
	4	15652	11.55
	8	16700	12.57

SPENT PWR FUEL TYPE	BS	PEAK CHANNEL POWER (MW)	PEAK BUNDLE POWER (KW)	CPPF
Nat. U	2	7252 (J11)	868 (E12,6)	1.021 (P18)
	4	7231 (J11)	839 (E12,6)	1.038 (V12)
	8	6991 (H8)	837 (E12,6)	1.079 (E16)
3.2 w/o 35 GWD/Te(U)	2	6745 (G15)	766 (J14,3)	1.049 (P18)
	4	6731 (F15)	792 (H15,4)	1.102 (P18)
	8	6889 (H15)	856 (E12,6)	1.243 (P18)
3.2 w/o 30 GWD/Te(U)	2	6872 (E9)	781 (E14,4)	1.067 (N17)
	4	6765 (F15)	830 (F15,4)	1.141 (P18)
	8	6872 (H8)	863 (H7,7)	1.357 (P18)
3.5 w/o 35 GWD/Te(U)	2	6572 (F15)	819 (F15,4)	1.062 (P18)
	4	6846 (E14)	778 (F15,4)	1.131 (N17)
	8	6878 (H15)	861 (H16,6)	1.326 (P18)

DUPIC Results

- reference DUPIC cycle similar to SEU 0.9

- initial nuclide field in DUPIC sensitive to:

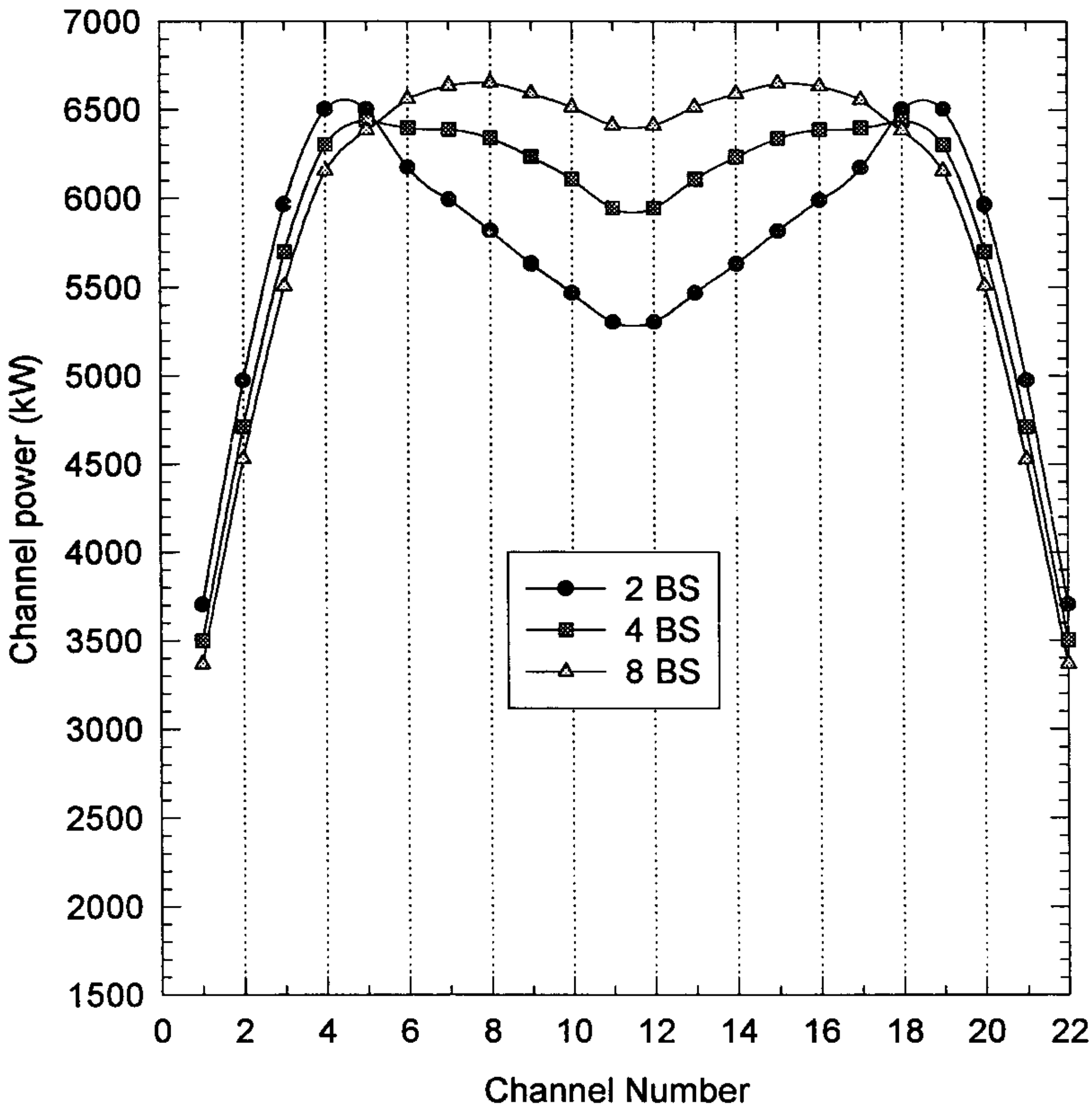
  - o PWR assembly burnup: $\frac{\Delta B_{DUPIC}}{\Delta B_{PWR}} \approx -1$
  - o PWR local parameters
  - o PWR initial enrichment (same burnup)

- with 2BS, CPPF is smaller than with Natural uranium cycle

- axial power shape similar to SEU

- other considerations:

  - o CANDU local parameters
  - o void effect (influence of B.P.)
  - o xenon
  - o control rod worth
  - o heterogeneity : uncertainty in CPPF due to uncertainty in initial nuclide field.



Radial Channel Power Profiles
(DUPIC, 3.~~5~~2 w/o, 30 GWD/Te(U), Row L)

Conclusion

- increasing the fuel enrichment in CANDU:
  - increases burnup and reduces fueling costs
  - reduce bundle scheme to maintain CPPF within current margins
  - fueling machine utilization (refueling frequency)
- problem with 2BS and high enrichment can easily be solved:
  - relocating of rows of adjusters rods (away from central positions) will:
    - flatten axial shape and reduce peak bundle power
    - increase reactivity worth of adjusters
    - reduce avg. exit burnup (increased leakage and absorption)
  - for reactors without adjusters, use a checkerboard pattern alternating 6BS and 2BS.
- fuel management optimization requires simultaneous optimization of control poisons (adjuster grading and layout in this case)
- flexibility of CANDU fuel cycles:
  - enriched uranium
  - recovered uranium
  - Pu disposal (MOX)
  - PWR spent fuel (DUPIC)
  - thorium fuel